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INTERNATIONAL WORKSHOP ON AGE-RELATED DEGRADATION OF REACTOR VESSELS AND INTERNALS

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Research Information Letter
Office of Nuclear Regulatory Research

Disclaimer

ABSTRACT

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research and Office of Nuclear Reactor Regulation organized this International Workshop on Age-Related Degradation of Reactor Vessels and Internals, held May 23–24, 2019, at NRC Headquarters, 11545 Rockville Pike, Rockville, MD.

Approximately 70 attendees from 31 regulatory, research, and industry organizations representing 11 countries attended and presented information on their organizations' state of knowledge, operating experience, and research activities related to age-related degradation of reactor pressure vessels and reactor vessel internals. The staff affirmed that aging management programs and research programs presented by international organizations are consistent and are evaluating similar issues as reactors enter periods of extended operation.

The objectives for this public workshop were to explore, via presentations representative of participating organizations from the United States, Japan, Czech Republic, Belgium, South Korea, United Kingdom, Hungary, France, and Switzerland, the following issues:

- state of knowledge, operating experience, and research activities related to reactor pressure vessel embrittlement at high fluence levels
- degradation of reactor vessel internals for operating periods beyond the original design up to 80 years
- degradation related to long-term operation of other safety-significant primary pressure boundary components

During the workshop and in subsequent correspondence, the workshop participants addressed four questions:

(1) (a) What program or guidance forms the basis of your aging management approach?

(b) What do you believe are the most significant technical issues related to long-term operation?

(c) Does your country have any plans to update its regulatory guidance to address any of these aging management issues?

(2) Radiation-induced void swelling and creep are degradation mechanisms of potential concern during long-term operation. Summarize or reference any operating experience, research programs, or aging management programs related to void swelling or creep.

(3) (a) What is your current embrittlement trend curve (ETC) for reactor pressure vessel steels, and when was it implemented into your regulations?

(b) Does your country use ETCs for predictive purposes, or do you rely on surveillance results and then use ETCs for interpolating among surveillance results?

(4) Does your country have any plans to use additive manufacturing (or other advanced manufacturing techniques) for repair/replacement of components?

FOREWORD

The Atomic Energy Act of 1954, as amended, and U.S. Nuclear Regulatory Commission (NRC) regulations limit commercial power reactor licenses to an initial 40 years but also permit such licenses to be renewed. This original 40-year term for reactor licenses was based on economic and antitrust considerations—not on limitations of nuclear technology. Due to this selected period, however, some structures and components may have been engineered on the basis of an expected 40-year service life. In 1982, the NRC established a comprehensive program for nuclear plant aging research. The results of this research indicated that most nuclear plant aging issues are manageable and do not pose technical impediments that would prevent them from operating for additional years beyond their original 40-year license period.

The NRC Office of Nuclear Regulatory Research provides specific research products to facilitate the evaluation of aging effects on passive long-lived systems, structures, and components (SSCs). The objective of this research is to generate independent technical data and confirmatory tools to enable development of regulatory guidance on the aging of SSCs and to support the regulatory review of future applications for license renewal. These products build on analysis methods, tools, and expertise developed as part of ongoing and new research activities, focused specifically on aging effects during long-term operation.

These new research activities include NRC and industry public workshops. These workshops are intended to address the state of knowledge on the technical issues identified in the staff requirements memorandum dated August 29, 2014, for SECY-14-0016, “Ongoing Staff Activities to Assess Regulatory Considerations for Power Reactor Subsequent License Renewal,” dated January 31, 2014, and discussed in NUREG-2191, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” issued July 2017, and any new operating experience from the initial license renewal period.

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ABBREVIATIONS AND ACRONYMS

Term	Description
ADAMS	Agencywide Documents Access and Management System
AIC	Ag-In-Cd surface treatment, in the context of ion-nitride treated RCCAs
AM	aging management
AMP	aging management program
AMRC	Advanced Manufacturing Research Centre, University of Sheffield (U.K.)
AMT	advanced manufacturing technologies
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (France)
ASTM	ASTM International (formerly American Society for Testing and Materials)
Bel V	technical support organization (Belgium)
BWR	boiling-water reactor
BWRVIP	BWR Vessels and Internals Program
CANDU	Canada Deuterium Uranium (reactor)
CE	Combustion Engineering
CEZ	České Energetické Závody (Czech public electric utility)
CFR	Code of Federal Regulations
CGR	crack growth rate
CRDM	control rod drive mechanism
CRIEPI	Central Research Institute of Electric Power Industry (Japan)
CT	computed tension
CVN	Charpy V-notch
DE	Division of Engineering, in NRC/RES
DETEC	Department of the Environment, Transport, Energy, and Communications (Switzerland)
DMW	dissimilar metal welds
DNRL	Division of New and Renewed Licenses, in NRC/NRR
DOE	U.S. Department of Energy
EAC	environmentally assisted cracking
EAD	engineering principles for aging and degradation
EAF	environmentally assisted fatigue
EDF	Électricité de France
ENSI	Swiss Federal Nuclear Safety Inspectorate
ENSREG	European Nuclear Safety Regulator’s Group
EPRI	Electric Power Research Institute
ETC	embrittlement trend curve
FAC	flow-accelerated corrosion
GALL	Generic Aging Lessons Learned, NUREG-1801
GCWM	guide card wear measurement
HCF	high-cycle fatigue
IAEA	International Atomic Energy Agency
IASCC	irradiation-assisted stress-corrosion cracking
IDOM	Ingeniería y Dirección de Obras y Montaje (Spain)

Term	Description
IGALL	International Generic Aging Lessons Learned
IMT	issues management tables (EPRI)
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (France)
ISI	in-service inspection
JAEA	Japan Atomic Energy Agency
JEPIC	Japan Electric Power Information Center, U.S.A.
KAERI	Korea Atomic Energy Research Institute
KATAM	catalogue of potential aging mechanisms
KHNP	Korea Hydro & Nuclear Power Co., Ltd
KINS	Korea Institute of Nuclear Safety
KKL	Kernkraftwerk Leibstadt Nuclear Power Plant (Switzerland)
KKM	Muhleberg Nuclear Power Plant (Switzerland)
KSNP	Korean Standard Nuclear Plant
LBP	late blooming phases
LCF	low-cycle fatigue
LEA	limited embrittlement area
LRGD	license renewal guidance document
LTO	long-term operation
LWR	light-water reactor
LWRS	Light Water Reactor Sustainability (Program)
MAPC	Materials Action Plan Committee (EPRI)
MAWG	Materials Assessment Working Group (EPRI)
MDM	materials degradation matrix (EPRI)
MEM	maintenance effectiveness monitoring
Mn	manganese
MPC	Materials Performance Centre, University of Manchester (U.K.)
MRP	Materials Research Pathway (part of LWRS Program)
MSIP	Mechanical Stress Improvement Process
MTR BR2	material test reactor BR2 (Belgium)
MVM Paks	Magyar VillamosMűvek, NPP in town of Paks, Hungary
NDE	nondestructive examination
NEI	Nuclear Energy Institute
Ni	nickel
NPP	nuclear power plant
NRA	Nuclear Regulatory Authority (Japan)
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (in NRC)
NSSC	Nuclear Safety and Security Commission (South Korea)
NTD ASI	Normative Technical Documentation of Czech Association of Mechanical Engineers
ODSCC	outer diameter stress-corrosion cracking
OE	operating experience
ONR	Office for Nuclear Regulation (U.K.)
OpEx	operating experience

Term	**Description**
ORNL	Oak Ridge National Laboratory
PCVN	pre-cracked Charpy V-notch
PFM	probabilistic fracture mechanics
PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWROG	Pressurized-Water Reactor Owners' Group
PWSCC	primary water stress-corrosion cracking
R&D	research and development
RCCA	rod cluster control assembly
RCS	reactor cooling system
RES	Office of Nuclear Regulatory Research (in NRC)
RG	regulatory guide
RI	reactor internal
RPV	reactor pressure vessel
RR	repair and replacement
RT_{NDT}	Reference Temperature for Nil Ductility Transition.
RV	reactor vessel
RVI	reactor vessel internal
SALTO	Safety Aspects of Long Term Operation
S/NRA/R	Division of Research for Reactor System Safety, Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (Japan)
SCC	stress-corrosion cracking
SCK CEN	Studiecentrum voor Kernenergie; Centre d'Étude de l'énergie Nucléaire (Belgium)
SLR	subsequent license renewal
SSC	systems, structures, and components
SUJB	State Office for Nuclear Safety (Czech Republic)
TF	thermal fatigue
TLAA	time-limited aging analysis
TMF	thermomechanical fatigue
TTS	transition temperature shift
TWCF	through-wall crack frequency
UJV Rez	Nuclear Research Institute Rez (Czech Republic)
U.K.	United Kingdom
UPI	upper plenum injection
UT	ultrasonic technology
VT	visual testing
VVER (WWER)	water-water energetic reactor, series of pressurized-water reactor designs originally developed in the Soviet Union
WEC	Westinghouse Electric Corporation
WOL	weld overlay

1 INTRODUCTION

This research information letter summarizes the presentations and discussions at an international workshop held on May 23–24, 2019, at the U.S. Nuclear Regulatory Commission (NRC) Headquarters auditorium in Rockville, MD (Figures 1 and 2). Participants from 31 regulatory, research, and industry organizations representing 11 countries (Table 1) took part in presentations and discussions on the state of knowledge, operating experience (OE), and research activities related to reactor pressure vessel (RPV) embrittlement at high fluence levels, degradation of reactor vessel internals (RVIs) for operating periods beyond the original design up to 80 years, and degradation related to long-term operation (LTO) of other safety-significant primary pressure boundary components.

1.1 Participating Organizations

Table 1-1 lists the organizations that participated in the workshop.

Table 1-1. Participating Organizations

Organization	Country	Link
Arizona Public Service	U.S.A.	http://aps.com
ASN	France	http://www.french-nuclear-safety.fr/
Bel V	Belgium	https://www.belv.be/index.php/en/
CEZ	Czech Republic	https://www.cez.cz/en/home
CRIEPI	Japan	https://criepi.denken.or.jp/en/
EDF	France	https://www.edfenergy.com/
ENSI	Switzerland	https://www.ensi.ch/en/
Entergy	U.S.A.	https://www.entergy.com/
EPRI	U.S.A.	https://www.epri.com/
Exelon	U.S.A.	https://www.exeloncorp.com/
Framatome	France	https://www.framatome.com/EN/home-57/index.html
IAEA	Austria	https://www.iaea.org/
IDOM	Spain	https://www.idom.com/
IRSN	France	https://www.irsn.fr/EN/Pages/Home.aspx
JEPIC-USA	U.S.A.	https://www.jepic-usa.org/
KAERI	South Korea	https://www.kaeri.re.kr/eng/
KHNP	South Korea	http://www.khnp.co.kr/eng/main.do
KINS	South Korea	https://www.kins.re.kr/en/
KKL	Switzerland	https://www.kkl.ch
MVM Paks	Hungary	http://www.atomeromu.hu/hu/Lapok/default.aspx
NRA	Japan	https://www.nsr.go.jp/english/
NRC	U.S.A.	https://www.nrc.gov/
ONR	U.K.	http://www.onr.org.uk/
ORNL	U.S.A.	https://www.ornl.gov/
Ringhals AB	Sweden	https://group.vattenfall.com/se/var-verksamhet/ringhals
SCK CEN	Belgium	https://www.sckcen.be/en

Organization	Country	Link
Southern Nuclear Company	U.S.A.	https://www.southerncompany.com/our-companies/southern-nuclear.html
SUJB	Czech Republic	https://www.sujb.cz/en/
Tractebel	Belgium	https://tractebel-engie.com/en
UJV Rez	Czech Republic	https://www.ujv.cz/en
University of Manchester MPC	U.K.	https://www.materials.manchester.ac.uk/materials-performance-centre/

This document includes the presentation slides, along with brief summaries of the workshop presentations. The workshop attendees are listed in Appendix A.

The views and opinions presented in this report are those of the individual participants, and publication of this report does not constitute NRC approval or agreement with the information contained herein. As such, these proceedings are not a substitute for NRC regulations or guidance. Rather, the approaches and methods described in these proceedings and the recommendations from the discussions are provided for information only, and compliance is not required. Use of product or trade names in this report is for identification purposes only and does not constitute endorsement by the NRC.

Figure 1 - Allen Hiser, U.S.A., speaking about the NRC's aging and materials research activities (see presentation 2.1)

Figure 2 - Kensaku Arai, Japan, presenting a flow chart for RPV integrity evaluation (see presentation 2.5)

1.2 Workshop Focus and Agenda

For the reader's reference, Table 1-2 indicates the represented countries, the presenters, their company or agency, and the general topic(s) on which they presented. The cross-hatch indicates presentations that were very focused and provided an in-depth discussion of the given topic. The solid grey shading indicates a more cursory discussion of that topic. Table 1-3 presents the formal workshop agenda.

Table 1-2. Focus Areas in Presentations Concerning Age-Related Degradation in Reactor Vessels and Internals

Country	Organization/ Speaker	Aging management guidance	Materials aging research activities	LTO concerns, operating experience	Irradiation effects, void swelling, creep	RPV embrittlement, ETC curves
U.S.A.	NRC/NRR (A. Hiser)	▨		░		░
U.S.A.	NRC/RES (R. Tregoning)		░			
U.S.A.	EPRI (M. Burke)		▨	░		
U.S.A.	ORNL (F. Chen)		▨			░
Japan	NRA (K. Arai)		▨			▨
Japan	CRIEPI (T. Arai)				░	▨
Czech Republic	CEZ (J. Ertl)	▨				░
Czech Republic	UJV (M. Zamboch)		▨		▨	░
Belgium	SCK CEN (S. Gavrilov)		░			
Belgium	Tractebel (C. Dupuit)			░		
Belgium	SCK CEN (M. Lambrecht)					▨
Belgium	Tractebel (M. De Smet)			░		
Belgium	Tractebel (C. Dupuit)			░		
South Korea	KINS (T-K Song)	▨				
South Korea	KAERI (B-S Lee)		▨	░		▨
South Korea	KHNP (J-S Yang)					
U.K.	ONR (G. Hopkin)	░				░
U.K.	University of Manchester (G. Burke)		▨		▨	▨
Hungary	MVM Paks NPP (S. Ratkai)	▨	░		░	
France	EDF (R. Menand)	░		░	░	
France	IRSN (E. Viard)			░		
Switzerland	ENSI (R. Doering)	▨	▨	░		░
Switzerland	KKL (J. Heldt)			░		

Table 1-3. Workshop Agenda

Day 1 (May 23, 2019)	
Time	**Presentation Topic**
0800	Welcome/Introductions/Logistics
0815	Country Presentation, United States (NRC)
0900	Country Presentation, United States (EPRI)
1000	Break
1015	Country Presentation, Japan
1130	Lunch
1300	Country Presentation, Czech Republic
1415	Country Presentation, Belgium
1530	Break
1545	Panel Discussion
1645	Public Comments
1645	Adjourn
Day 2 (May 24, 2019)	
Time	**Presentation Topic**
0800	Introduction
0815	Country Presentation, South Korea
0930	Country Presentation, United Kingdom
1045	Break
1100	Country Presentation, Hungary
1215	Lunch
1330	Country Presentation, France
1445	Country Presentation, Switzerland
1600	Break
1615	Panel Discussion
1700	Public Comments
1715	Adjourn

2 SUMMARY OF PRESENTATIONS

On May 23–24, 2019, the NRC Office of Nuclear Regulatory Research (RES), Division of Engineering (DE), organized and hosted the International Workshop on Age-Related Degradation of Reactor Vessels and Internals. This workshop on the aging degradation of reactor vessels and internals included presentations (Figure 1) by senior staff at the NRC, as well as presentations by representatives from American and international industry, a U.S. Department of Energy (DOE) National Laboratory, and international nuclear regulators. All presentation materials are also publicly available in the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession No. ML19150A174.

The audience included approximately 70 attendees representing 31 companies and organizations from 11 countries, including industry groups, government regulatory agencies, and both foreign and domestic research institutes (see Section 1.1). Table 1 of this report gives links to the organizations represented by the speakers.

In the weeks following the workshop, interaction continued between the NRC staff and speakers to further clarify their perspective on key issues (see Section 3). In this section, presentation summaries are based on the author's post-meeting contribution, if any, and NRC staff notes from the workshop.

May 23, 2019—Morning Session

United States

2.1 Aging Management and Subsequent License Renewal in the United States (A. Hiser)

Allen Hiser, Senior Technical Advisor for License Renewal Aging Management in the NRC Office of Nuclear Reactor Regulation (NRR), Division of New and Renewed Licenses, spoke on aging management and subsequent license renewal (SLR) in the United States (ADAMS Accession No. ML19143A269) (Figure 1a). He gave an overview of the U.S. experience with license renewal and SLR and stressed that in the United States, the license renewal safety principles are that plant safety is assured by the regulatory process and requires additional actions for aging management of passive, long-lived structures and components during the license renewal period. He emphasized neutron embrittlement of the RPV and high-fluence effects on RVIs as being SLR technical concerns, and he included neutron embrittlement for RPV support steel elements as a new LTO technical issue arising from the review of SLR applications. In general, his concerns for aging were related to the ability to identify potential new aging phenomena.

2.2 NRC's Aging and Materials Research Activities (R. Tregoning)

Rob Tregoning, NRC Technical Advisor for Materials Engineering (RES/DE), next presented an overview of the NRC's work in materials and aging research (ADAMS Accession No. ML19150A186). He explained that research objectives in this area typically focus on supporting regulatory decision making related to the use of new materials, manufacturing technologies, and in-service inspection (ISI) techniques; address knowledge gaps related to materials degradation during long-term plant operation to 80 years; or inform and enhance the use of risk information in regulatory decision making. His overview summarized the NRC's research activities on primary water stress-corrosion cracking (PWSCC); irradiation-assisted stress-corrosion cracking (IASCC); steam generator tube integrity; aging management; spent fuel dry storage; neutron absorber materials; advanced manufacturing technologies; RPV integrity; piping integrity; probabilistic component integrity evaluation; and nondestructive examination (NDE). For each topic, he summarized the objective, motivation, and intended regulatory application; listed collaborative partners; identified recent accomplishments and deliverables; and discussed the next steps in the research. More information on each of these programs is available in NUREG-1925, Revision 4, "Research Activities: FY 2018–2020," issued March 2018 (ADAMS Accession No. ML18071A139).

2.3 U.S. Nuclear Electric Power Generation Industry Management of Age-Related Degradation (M. Burke)

Michael Burke, technical executive at the Electric Power Research Institute (EPRI), presented EPRI's systematic approach to the management of reactor components' aging degradation (ADAMS Accession No. ML19150A187). This approach has been employed by the U.S. nuclear power generation industry and supported by the formal processes of EPRI's Boiling-Water Reactor (BWR) Vessels and Internals Program (BWRVIP) and Materials Reliability Program since 2004. The presentation described how, building on the recognition of emerging aging issues from around 2000, the industry undertook a systematic approach to manage the mechanisms of materials aging degradation. The development of this systematic approach, based on the integrated issues management initiative outlined in Nuclear Energy Institute (NEI) 03-08,[1] led to the systematic categorization of aging degradation issues as a framework for aging management of the fleet. Dr. Burke described how the BWRVIP and Materials Reliability Program currently categorize aging management issues as assessment, inspection and evaluation, mitigation, repair and replacement, and regulatory-driven to identify, analyze, and assess the importance of, provide resources to, and resolve the key aging management issues on a timely basis. The process initially focused on U.S. industry issues under the aegis of NEI 03-08 but is now used for international members as well. The approach of collecting OE and identifying the most urgent issues in each category as the basis for systematic plant aging management via EPRI's Materials Degradation Matrix was described. The use of the issues management tables (IMTs) and their "Knowledge Gap" tables was described and discussed. Dr. Burke presented examples of how the IMT process has been employed to address and close knowledge gaps and to support successful management of age-related degradation of reactor vessels and internals.

2.4 Overview of Metals Research in LWRS Program MRP (F. Chen)

Frank Chen, senior scientist at Oak Ridge National Laboratory (ORNL), presented an overview of research on metals degradation in the Light Water Reactor Sustainability (LWRS) Program (http://lwrs.inl.gov) Materials Research Pathway (MRP) (ADAMS Accession No. ML19150A188). ORNL's research involves collaborations with partners including EPRI, Westinghouse, the Pressurized-Water Reactor Owners' Group (PWROG), Central Research Institute of Electric Power Industry (CRIEPI), Rolls Royce, Exelon, and the NRC. In describing the key benefit of the MRP research, Dr. Chen stated, "Understanding which components are susceptible to certain forms of degradation, and their predictive behavior, will permit more focused component inspections, component replacements, and more detailed regulatory guidelines." The LWRS MRP was illustrated by a Venn diagram including experimental testing, harvested materials, and modeling. Dr. Chen described elements of the MRP metal-related project portfolio in detail,

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[1] NEI 03-08, Revision 3, "Guideline for the Management of Materials Issues," February 2017, ADAMS Accession No. ML19079A256.

including RPV studies, mitigation technologies, harvesting and integrated research, and studies on core internals and piping. For example, ORNL's RPV studies have developed a testing technology for determining master curve fracture toughness for RPV steels using mini compact tension specimens machined from Charpy impact specimens.

Japan

2.5 Overview of Safety Research on Metal Aging Due to Neutron Irradiation in S/NRA/R (K. Arai)

Kensaku Arai, in the Division of Research for Reactor System Safety, Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R), presented an overview of safety research on metal aging due to neutron irradiation (Figure 1b) (ADAMS Accession No. ML19150A183). After summarizing previous research programs on this topic, he discussed current work on RPV embrittlement and probabilistic fracture mechanics (PFM). A PFM guidebook for RPVs has been developed that supports the calculation of probabilistic numerical indices such as through-wall crack frequency by PFM analyses. This presentation introduced the status of the safety research programs for neutron irradiation embrittlement and PFM that are currently being conducted by S/NRA/R.

2.6 CRIEPI Research Activities on Neutron Irradiation Embrittlement of RPV and Core Internals (T. Arai)

Taku Arai, CRIEPI, continued with another discussion of research activities on neutron irradiation embrittlement of RPV and core internals (ADAMS Accession No. ML19150A184). CRIEPI's research has focused on understanding the mechanism of embrittlement through correlating changes in microstructure and mechanical properties. CRIEPI proposes that Japan adopt a new embrittlement trend curve (ETC) based on high-fluence surveillance data and extensive atom probe tomography data. An ongoing confirmatory study on embrittlement of irradiated steels uses material harvested from the decommissioned Zion Unit 1. Another part of the research is investigating the combined effect of initial toughness distribution and fluence attenuation in RPV steel to evaluate conservatism in integrity assessment.

May 23, 2019—Afternoon Session

Czech Republic

2.7 Czech Approach to Ageing Management of Reactor Pressure Vessel and Reactor Vessel Internals—State of Knowledge (J. Ertl)

Jakub Ertl, CEZ, gave an overview of the Czech approach to the aging management of RPVs and RVIs (ADAMS Accession No. ML19150A180). He explained how the Czech Republic has two different types of aging management programs (AMPs)—component-based AMPs and specific AMPs (focused on the degradation mechanisms). Specific AMPs were created according to generic attributes prescribed by International Atomic Energy Agency (IAEA) recommendations (International Generic Aging Lessons Learned (IGALL[2])). He presented a

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[2] International Atomic Energy Agency, "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)," IAEA Safety Reports Series No. 82, Vienna, Austria, 2015.

new specific AMP, now in preparation, for radiation damage of internals. Another interesting document in preparation concerns RVI lifetime assessment.

2.8 UJV Activities in International Research Projects in the Field of RPV and RVI (M. Zamboch)

Miroslav Zamboch, UJV Rez, gave an overview of UJV activities in international RPV and reactor pressure internals research programs (ADAMS Accession No. ML19150A181). One such European research project is SOTERIA. The project's name is an abbreviation derived from "Safe lOng-TERm operation of light water reactors based on Improved understanding of radiation effects in nuclear structural mAterials." Other collaborative activities discussed included various European and international materials research programs. Each of these multinational research programs is focused on a different aspect of the aging, degradation, and integrity of RPV and RVI materials, such as environmental fatigue, radiation embrittlement, and probabilistic assessment of RPV integrity. The Czech Republic is performing a deterministic assessment of RPV embrittlement based on the results of the surveillance program Charpy impact tests and normative trend curves. The probabilistic approach to the assessment of RPV embrittlement using a master curve approach is performed as a voluntary, supplementary assessment.

2.9 Measurement of Core Shroud at NPP Temelin (M. Zamboch)

Miroslav Zamboch, UJV Rez, next discussed measurement of the core shroud at NPP Temelin, a VVER 1000 type, in operation since 2001 (ADAMS Accession No. ML19150A182). He emphasized that RPV internals of VVER 1000 are, because of operating conditions, sensitive to the development of irradiation-induced creep and swelling (resulting from high fluences, high doses, and high temperature due to gamma heating). At this moment, the evaluation of this degradation mechanism in RVIs is only computational in the Czech Republic. The development of the swelling may result in macroscopic change of the core shroud. For LTO (60+ years), it is useful to confirm and monitor the development of the swelling by measurement at the NPP directly. The measuring device able to detect specific changes of core shroud geometry was designed and developed at UJV Rez, and the methodology for measurement was prepared and approved by the NPP. The expected degradation mechanisms include radiation swelling and radiation creep, as well as fatigue, IASCC, wear, loss of fracture toughness, and mechanical damage. The presentation extensively discussed radiation swelling and radiation creep, including a basic explanation of the degradation mechanisms and methods to calculate their effects. UJV's work on measuring dimensional changes in the core shroud, beginning in 2021, is intended to provide real phenomenological information about any actual radiation swelling.

Belgium

2.10 Belgian R&D on Environmental Effects on Materials Degradation in LWRs (S. Gavrilov)

Serguei Gavrilov, SCK CEN, discussed Belgian research and development (R&D) on the effects of the environment on materials degradation in LWRs (ADAMS Accession No. ML19150A175). The presenter summarized results from IASCC studies and addressed hydrogen effects on RPVs in depth. Work continues on corrosion fatigue to address effects of surface conditions, hold time, and mean stress/strain.

2.11 Repair of Doel 1 NPP Reactor Vessel Head Penetrations (C. Dupuit)

Charles Dupuit, Tractebel, gave a detailed presentation of work done to repair the Belgian Doel 1 NPP reactor vessel head penetrations (ADAMS Accession No. ML19150A176). The Doel 1 plant is a two-loop pressurized-water reactor (PWR) connected to the grid in 1974. He emphasized that before Doel LTO, PWSCC indications on reactor vessel head penetrations were detected and dispositioned through a Justification of Continuous Operation. After the LTO decision, plans for a vessel head replacement were deemed not feasible, and a repair of vessel head penetrations was therefore mandatory. This presentation covered the repair scope, repair process (Inside Diameter Temper Bead), repair qualification and issues encountered, as well as the ISI program for later outages.

2.12 Belgian R&D Using the Enhanced Surveillance Strategy for RPV Embrittlement Assessment (M. Lambrecht)

Marlies Lambrecht, SCK CEN, presented Belgian research to assess RPV embrittlement via an enhanced surveillance strategy (ADAMS Accession No. ML19150A177). The enhanced surveillance supplements the mandatory conventional surveillance, based on Charpy impact tests, and includes tensile testing with sub-sized specimens, facilitating radiation damage modeling. Researchers at SCK CEN are continuously developing and improving the ETC model with the support of data from the high-performance material test reactor BR2 and reliable databases. This work should provide increased reliability in the surveillance database in the context of LTO.

2.13 Doel 1 & 2 Upper Plenum Injection Line Issue (M. De Smet)

Michel De Smet, Tractebel, presented OE associated with the Doel 1 and 2 upper plenum injection (UPI) lines, part of the safety injection system and typical for a Westinghouse two-loop PWR (ADAMS Accession No. ML19150A178). He noted that in April 2018, a leak occurred in one of the two UPI lines of Doel 1 NPP. NDE of the UPI lines revealed degradation in the longer UPI-A lines of Doel 1 and 2 NPP: cracking in the bottom part of the straight pipe, upstream of the weld between straight pipe and elbow, and circumferential cracking in the weld between straight pipe and elbow. No cracking was found in the shorter UPI-B lines. Destructive examinations performed in two independent laboratories confirmed that the degradation was due to thermal fatigue. The affected straight parts of the UPI-A lines were replaced. A safety demonstration and justification for safe restart of both units, relying on repair, comprehensive monitoring, and future inspections, were submitted and were approved by the Belgian safety authorities. The installed monitoring confirms the presence of thermal cycles in the UPI lines at full-power conditions. Structural integrity evaluations (stress, fatigue, and fatigue crack growth) are ongoing.

2.14 Inspection of Control Rod Guide Assemblies in Belgian NPPs (C. Dupuit)

Charles Dupuit, Tractebel, described his experience inspecting control rod guide assemblies in Belgian PWRs, including Tihange 3 and Doel 4 (ADAMS Accession No. ML19150A179). He noted that following inspections in accordance with international guidelines and recommendations, control rod guide card wear was observed in most Belgian units. Excessive wear was detected at Tihange 3. As this unit shares many similarities with Doel 4, where no excessive wear was present, the wear mechanism was studied. Tractebel hypothesized that the difference in observed wear is possibly related to the different length of ion nitride (AIC) coating present in the rod cluster control assemblies (RCCAs) at Tihange 3 versus those at Doel 4. Mr. Dupuit also presented an overview of similar inspections of other units.

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South Korea

2.15 Current Status of Aging Management on Reactor Vessels in Korea (focusing on surveillance test) (T-K Song)

Tae-Kwang Song, KINS, discussed the current status of aging management of reactor vessels in South Korea, focusing on surveillance test requirements (ADAMS Accession No. ML19150A193). He noted that a surveillance test has been performed to monitor changes in the fracture toughness properties of reactor vessel materials. Two issues in surveillance testing were introduced: (1) the low value of lead factor and (2) the 60-year design life. The surveillance capsule of the KSNP (Korean Standard Nuclear Plant) type reactor is located on the inside wall of the reactor vessel, and thus, the lead factor of a KSNP is less than that of a Westinghouse-type reactor. To overcome the problems caused by low lead factors at KSNP, the capsule withdrawal schedule was adjusted to obtain irradiation data at the end of licensed operation (i.e., 40 years). New surveillance capsules were fabricated using archive materials of KSNP reactors, then inserted into Westinghouse-type reactors, making use of their higher lead factor to obtain irradiation data at the extended life (60 years). The current surveillance test requirement is based on a 40-year design life as noted in Korean Nuclear Safety and Security Commission (NSSC) Notice 2017-20 and ASTM E185-82. Thus, it is difficult to apply the current requirement directly to the recently developed power plant with a 60-year design life. A draft bill of surveillance requirement to resolve this problem has been developed so that it can be applied to all reactors regardless of design life. This bill is in the process of rulemaking in South Korea.

2.16 Ongoing Researches in Age-Related Degradation of Reactor Materials in Korea (B-S Lee)

Bong-Sang Lee, KAERI, discussed ongoing research in age-related degradation of reactor materials in South Korea (ADAMS Accession No. ML19150A194). He noted that irradiation embrittlement of a high copper weld at higher neutron dose was characterized with surveillance test data from both Charpy and pre-cracked Charpy V-notch (PCVN) specimens. The transition shifts from PCVN fracture toughness data were almost the same as Charpy V-notch impact data for the high copper weld. High-flux experimental data obtained from research reactors were comparable to those from surveillance tests of Korean RPV steels. ETC models might have underpredicted the measured transition temperature shift (TTS) values for Korean RPV steels and welds, especially at higher fluence regions. Baffle-former bolts in the Kori-1 PWR have shown some defect indications from the final ISI signals. Those bolts will be extracted and investigated to identify a possible IASCC mechanism. Furthermore, the retired Kori-1 reactor components are going to be used for aging degradation studies under nuclear safety R&D projects. Advanced NDE technology and corrosion-related research are also important topics, with projects underway to improve the knowledge of NPP aging management.

2.17 Operating Experience (OpE) on RV Internals, RV Head Penetrations, and RCS Small Bore Nozzles in Korea (J-S Yang)

Jun-Seog Yang, KHNP, presented OE of reactor head, reactor internals, and small-bore nozzles attached at the reactor coolant pipe (ADAMS Accession No. ML19150A195). He noted that in 2012, at Hanbit Unit 3, a Combustion Engineering (CE) two-loop plant, ultrasonic technology (UT) detected flaws in six control rod drive mechanism (CRDM) penetrations. All the reported flaws were oriented axially and located in the CRDM nozzles near the location of the J-groove weld toe. Reactor vessel (RV) heads of Hanbit Units 3 and 4 were replaced with Alloy 690/52/152 heads. Welds of RV head penetrations of Hanwul Units 3 and 4 and Hanbit Units 5 and 6 are scheduled to be overlay-welded with 52/152 weld metal. In 2015, Kori Unit 1, a Westinghouse Electric Corporation (WEC) two-loop plant, reported indications of cracking in baffle-former bolts for the first time in South Korea. In 2016, at Hanwul Unit 3, a CE two-loop plant, a leak was discovered on the sampling nozzle of reactor cooling system (RCS) hot-leg piping. The half-nozzle repair technique will be used on small-bore nozzles attached to RCS hot-leg piping and pressurizer heater sleeves. The Korean regulatory body required the utility to identify the root cause of bolt damage and to establish plans for the inspection and evaluation of RVIs to manage aging effects.

United Kingdom

2.18 UK Regulatory Experience in Materials Ageing (G. Hopkin)

Gareth Hopkin, Office for Nuclear Regulation (ONR), gave an overview of the United Kingdom (U.K.) regulatory experience in materials aging (ADAMS Accession No. ML19150A198). Starting with a history of nuclear power in the United Kingdom, he progressed to ONR safety assessment principles, which include five different engineering principles for aging and degradation (EAD). For example, EAD.03 relates directly to materials aging, including irradiation embrittlement. He emphasized that the U.K. regulatory approach is goal setting and not prescriptive. The management of materials aging is to be addressed as part of the generic design assessment process. The ONR expectation for irradiation embrittlement of major vessels is that code compliance is not necessarily sufficient. For example, the United Kingdom's one civil LWR, Sizewell B, has a surveillance program containing Charpy and fracture toughness specimens, including pre-strained samples. For new builds, the expectations vary between technologies and reactor designs, depending on the nuclear safety significance and the importance of the degradation mechanism to overall integrity. New build reactors in the United Kingdom must demonstrate that they have suitable and sufficient surveillance programs. The requesting party must demonstrate an adequate understanding of the mechanism of irradiation embrittlement and how the specifics of the reactor design interact with this mechanism.

2.19 State of Knowledge and Research Activities on RPV Materials in UK (G. Burke)

Grace Burke, Director of the Materials Performance Centre (MPC) at the University of Manchester, gave an in-depth tutorial on the evolution of current understanding of irradiation embrittlement of RPV materials (ADAMS Accession No. ML19150A199). She included studies on the properties and physical changes in neutron-irradiated steels and welds and emphasized model development (based on empirical and fundamental understanding) for predicting behavior. She also discussed research activities in the United Kingdom, focusing on irradiation-induced degradation of structural materials. She stated that there is no need to invoke late-blooming phases to explain irradiation behavior in RPV steels and welds; rather, irradiation embrittlement can be explained by the evolution of solute-enriched clusters. She mentioned domestic and international collaborations that contribute to ongoing and proposed research. The U.K. National Nuclear Users Facilities, for example, provide government-supported nuclear R&D. In addition to gathering data from real (harvested) components, she proposed to pursue continued collaboration with the International Group on Radiation Damage Mechanisms, originally started with U.S. NRC guidance.

Hungary

2.20 AM and LTO-Related Activities of RPV and Its Internals and Other Primary Pressure Boundary Components at the Paks NPP (S. Ratkai)

Sandor Ratkai, of the MVM Paks NPP, discussed aging management and LTO-related activities for RPVs, internals, and other primary pressure boundary components (ADAMS Accession No. ML19150A191). AMPs for Hungary's VVER NPP are based on the NRC license renewal guidance documents, are updated periodically in accordance with IGALL, and take into account R&D results and U.S. and European OE. About 150 very detailed AMPs were developed for passive safety-related components, while active components are managed by a maintenance effectiveness monitoring process. Time-limited aging analyses (TLAAs) are also used in AMPs, and Mr. Ratkai described several examples. A new TLAA is being developed for void swelling and IASCC in RPV internals.

May 24, 2019—Afternoon Session

France

2.21 EDF Operating Experience RV Internals (R. Menand)

Roch Menand, Électricité de France (EDF), recalled EDF OE for RVIs, including wear of thermal sleeves (ADAMS Accession No. ML19150A189). Maintenance operations on components including guide tubes, split pins, CRDMs, and baffle bolts were used to illustrate the context for the EDF lifetime management policy and readiness for plant life extension. Next, he presented ongoing research on RVIs, such as flux thimble tubes, thermal sleeves, and baffle bolting. Research on irradiation of RVIs includes an investigation to determine PWR conditions that can lead to void swelling.

2.22 Carbon Segregations in Heavy Forged Components (E. Viard)

Emmanuel Viard, of the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), gave an overview of OE with carbon segregation in heavy forged components, as an assessment of nonconformance in some French PWR primary coolant systems (ADAMS Accession No. ML19150A190). Carbon contents exceeding maximum allowable values have been identified as the leading cause for low toughness values in low-alloy steel forgings. In response to the nonconformances, the French safety authority demanded a determination of the toughness of the segregated areas based on characterization of sacrificial components. For the future, Mr. Viard recommends that material quality demonstrations include drop weight tests, in addition to Charpy V-notch and compact tension (CT) fracture toughness tests. Additional conservatism on material properties is recommended to ensure that specimens are sufficiently representative, and that variability is addressed.

Switzerland

2.23 Aging Management and LTO of NPPs in Switzerland: Status 2019 (R. Doering)

Ralph Doering of ENSI shared detailed OE related to RPV inclusions (Beznau NPP), core shroud cracking (Muhleberg NPP (KKM)), fatigue, and embrittlement at various Swiss NPPs (ADAMS Accession No. ML19150A196). Results of RPV fatigue monitoring in the form of current fatigue usage factors and the corresponding levels extrapolated to 60 years of operation indicate that the LTO of Swiss NPPs is not subject to any limitations as a result of RPV material fatigue. Similarly, results of surveillance programs in the form of fluence calculations and the corresponding levels of irradiation embrittlement, extrapolated to 60 years of operation, show that the LTO of Swiss NPPs is not subject to any limitations as a result of RPV irradiation embrittlement. Switzerland established a systematic aging management methodology in 1991. The degradation mechanisms of concern include fatigue (low-cycle fatigue, high-cycle fatigue, thermal fatigue, thermomechanical fatigue, environmentally assisted fatigue (EAF)), irradiation embrittlement, stress-corrosion cracking (SCC), flow-accelerated corrosion (FAC), and others (corrosion, wear, erosion, thermal aging). Based on results of the extensive inspection programs, specific corrective actions have been taken for identified aging-related damage and degradation. With application of the AMP for RPVs, safe LTO for an operating period of 60 years (or, for KKM, until the final shutdown in December 2019) is ensured.

2.24 Operating Experience of a Swiss BWR (J. Heldt)

Jens Heldt, Kernkraftwerk Leibstadt NPP (KKL), shared detailed OE related to the Leibstadt NPP, especially associated with environmentally assisted cracking (EAC) observed in reactor water piping and SCC of dissimilar metal welds (ADAMS Accession No. ML19150A197). After weld overlay repairs of these cracks, fracture toughness tests and analysis of SCC susceptibility were conducted. A pilot application for a mechanical stress improvement process was prepared for the Swiss regulator. Mr. Heldt concluded by stating that the following are important for the assessment of SCC and EAC: understanding of the mechanisms and phenomena, disposition lines for crack growth, operational experience, fabrication history, and NDE capability.

3 DISCUSSION OF KEY ISSUES

During the workshop and in subsequent correspondence, the speakers were asked the following questions, with a request that they provide perspectives from their country:

(1) (a) What program or guidance forms the basis of your aging management approach (e.g., GALL, IGALL, other references)?

(b) What do you believe are the most significant technical issues related to long-term operation (LTO)? (Provide a list below and, if appropriate, applicable references.)

(c) Does your country have any plans to update its regulatory guidance to address any of these aging management issues?

(2) Radiation-induced void swelling and creep are degradation mechanisms of potential concern during LTO. Could you briefly summarize or reference any operating experience, research programs, or aging management programs related to void swelling or creep?

(3) (a) What is your current embrittlement trend curve (ETC) for reactor pressure vessel steels, and when was it implemented into your regulations?

(b) Does your country use ETCs for predictive purposes, or do you rely on surveillance results and then use ETCs for interpolating between surveillance results?

(4) Does your country have any plans to use additive manufacturing (or other advanced manufacturing techniques) for repair/replacement of components? (Provide brief summary below and if appropriate applicable references.)

The NRC staff compiled the answers that were received into Tables 4–7 below.

Table 3-1 captures verbatim the answers, received by e-mail from each participating country, for Questions 1 (a, b, and c) regarding aging management .

Table 3-1. Aging Management Approach

Country		Responses to Questions 1 (a, b, and c)
Czech Republic	(a)	Aging Management (AM) is based upon methodology elaborated by IAEA. The general approach of IAEA to AM represented by Safety Standard, Safety Fundamentals and Safety Requirements and consequently safety Guides was transformed into CEZ control documentation (documentation concerned with ageing management). The generic approach to AM is defined in Specific Safety Requirements No. SSR 2/2 Safety of Nuclear Power Plants: Commissioning and Operation. The guidance of Specific Safety Guide No. SSG-48: Ageing Management and Development of Programme for Long Term operation of Nuclear Power Plant (and its predecessors) was used for implementation of ČEZ AMPs. The compliance of AMPs with general world practice is assured by participation in international project as IGALL (International Generic Aging Lessons Learned—IAEA), benchmarking performed by European Nuclear Safety Regulator's Group ENSREG, regular information exchange between operators of VVER NPPs and experts from supporting technical organisations (Czech Republic, Slovakia, Hungary) during regularly performed workshops. The

Country		Responses to Questions 1 (a, b, and c)
		information from ongoing international research projects with possible connection to aging of reactor pressure vessel are evaluated on an annual basis. The AM process was reviewed several times during SALTO (Safety Aspects of Long-Term Operation) missions by IAEA experts. Evaluation of reactor pressure vessel integrity (with respect to specific degradation mechanisms is performed according to the NTD ASI (Normative Technical Documentation of the Czech Association of Mechanical Engineers), version 2017. The first edition of NTD ASI was issued in1996. It was based mainly upon Russian standard PNAE-G-7-002-86 and it represented first step of harmonisation with ASME approach. The next important modernisation of NTD ASI took place in 2008 edition where the VERLIFE 2003—“Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs during Operation” was developed within the 5th Framework Program of the European Union and incorporated into Czech technical standards. By that edition the new development in fracture mechanics and approaches of PWR codes were included together with IAEA PTS guidelines. In the 2013 the NTD ASI was upgraded according to outputs of 6th Framework Program “COVERS—Safety of WWER NPPs” of the European Union in 2008—updated version of VERLIFE 2003. At this moment the further NTD ASI development is ongoing based on results from “IAEA NULIFE—Plant Life Management of NPPs” project, where the VERLIFE procedure was further modernised and extended with collaboration of experts of PWR operating countries as well as of VVER operating countries including Russian experts. The next edition of NTD ASI is prepared for issue in 2019.
	(b)	For LTO it is necessary to know response of the material to the long term operation loads (fatigue, environment, irradiation). To obtain proper data for evaluation and prediction of the passive structure and components state during whole operation it is necessary to perform experimental program using material aged during real operation of the NPP. The most significant technical issue is change of the RVI material properties during long term operation (IASCC, RI swelling and creep, LEA formation). The other issue could be long term temperature aging for specific kind of steels. Degradation of concrete.
	(c)	In Czech Republic (with collaboration of the other countries) the project aiming for harvesting material (steel) from decommissioned unit Bohunice (Slovak Republic) has been started. Based upon material acquired the experimental program to provide information about in-situ aged components’ material shall be performed based on supporting funding.
Japan	(a)	Nuclear Regulation Authority (NRA) guidelines [see Table 8, below, for document references]: • The Guide for Extension of Operational Period on Commercial Power Reactors [1] • The Standard Review Plan for Extension of Operational Period on Commercial Power Reactors [2] • The Guideline on Implementing Measures for Aging Management at Commercial Nuclear Reactors [3] • The Standard Review Guideline for Aging Management at Commercial Nuclear Reactors [4] • Utility guidelines • Code on Implementation and Review of Nuclear Power Plant Ageing Management Program 2015 [5]

Country		Responses to Questions 1 (a, b, and c)
	(b)	NRA considers neutron irradiation embrittlement of reactor pressure vessels (RPVs), and aging of electrical penetration of primary containment vessels as long-term operation related research issues [6].
	(c)	NRA does not currently have plans to update the regulatory guidelines listed in 1(a).
South Korea	(a)	(The Korean NSSC Notice #2017-29 describes the requirements for scoping, screening, and assessment of aging management of Korean nuclear power plants, which refers to NUREG-1801 (GALL)). National R&D programs are also underway to improve the capabilities of aging management in addition to gathering the operating experiences.
	(b)	1) Cracking phenomena: PWSCC, IASCC, ODSCC 2) Environmental assisted fatigue analysis Design criteria at the time of construction may be different from the present design criteria. In that case, how to apply the current license basis to the existing plants for the long-term operation is important. Environmental assisted fatigue, for example, was not considered in the design status because it was not required at that time, however, it might be considered if it is re-evaluated for long-term operation 3) Integrity assessment of the embrittled components—RPV & RVI Effects of large seismic loadings on the embrittled RVI structure has been focusing on the integrity assurance of fuel elements.
	(c)	South Korea does not have a specific plan to update the regulatory guidance of aging management. However, several research projects are underway to improve technology of the aging management including the above issues.
Sweden	(a)	The basis is the GALL document
	(b)	1) Fracture toughness of the RPV beltline weld. Fracture toughness properties of the Stainless steels in the internals. Wear of internals, such as Lower Radial Support and Flux Thimble Tubes.
	(c)	The Swedish regulator SSM does not have any detailed regulatory guidance. It's up to the licensee to address these issues and justify adequate aging management and report it to the regulator.
Switzerland	(a)	Basis for AMP are the national laws (nuclear energy act, nuclear energy ordinance and DETEC ordinance "Preliminary shut down of NPP") and the IAEA NS-G-2.12/SSG-48. Other references are: • IAEA Technical Reports Series No. 448, Plant Life Management for Long Term Operation of Light Water Reactors, 2006. • IAEA-EBP-SALTO, Safety Aspects of Long Term Operation of Water Moderated Reactors, 2007. • IAEA Safety Reports Series No. 15, Implementation and Review of a Nuclear Power Plant Ageing Management Programme, 1999. • IAEA TecDocs No. 1361, 1470, 1471, 1556, 1557. EUR 22763, Development of a European Procedure for Assessment of High Cycle Thermal Fatigue in Light Water Reactors: Final Report of the NESC-Thermal Fatigue Project, 2007. Results by other programmes like iGALL and Operating Experience are taken into account.
	(b)	Most relevant, we expect, are embrittlement for the older reactors especially the PWR, fatigue including vibrational fatigue, SCC and FAC. Challenges are implementation of updated regulatory requirements (like increased hazards or safety margins).

Country		Responses to Questions 1 (a, b, and c)
	(c)	All currently relevant issues are addressed. AMP will [be] periodically reviewed according to the state-of-the-art and OpEx. The AMP-Guideline ENSI-B01 will be checked for the need of updates according to the new IAEA SSG-48 (2018) and the results of the ENSREG Topical Peer Review 2017.
U.S.A.	(a)	The guidance for aging management for license renewal (operation from 40 to 60 years) is provided in (as supplemented by license renewal-interim staff guidance documents (LR-ISG)): • Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Rev. 2), December 2010 • Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR, NUREG-1800, Rev. 2), December 2010 • The guidance for aging management for subsequent license renewal (operation from 60 to 80 years) is provided in (and will be supplemented by SLR-ISG documents) • Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report (NUREG-2191), July 2017 • Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants (SRP-SLR) (NUREG-2192), July 2017
	(b)	The most significant issues for SLR are: • Reactor pressure vessel neutron embrittlement at high fluence • Irradiation-assisted stress corrosion cracking of reactor internals and primary system components • Concrete and containment degradation • Electrical cable qualification, condition monitoring and assessment Material degradation issues are also periodically identified through operating experience. Some recent examples include selective leaching, control rod drive mechanism thermal sleeve degradation, and baffle former bolt failures.
	(c)	The NRC will update the SLR guidance for these topics and as needed for any other new information, using the SLR-ISG document process. Several SLR-ISGs are in process as of autumn 2020, related to mechanical, electrical and structures portions of the SLR guidance, and PWR RVI components (see https://www.nrc.gov/reading-rm/doc-collections/isg/license-renewal.html for more information on SLR-ISGs and LR-ISGs (the latter of which update the GALL report and the SRP-LR)).

Table 3-2, below, captures the responses from each country to Question 2, regarding radiation-induced void swelling and creep, degradation mechanisms of potential concern during LTO.

Table 3-2. Radiation-Induced Void Swelling and Creep

Country	Responses to Question 2
Czech Republic	Regular evaluation of the neutron irradiation load (fluxes, fluences) of the RVI material including assessment of the gamma heating (for each campaign)—included in the RPV and RVI AMP. Computational assessment of the RVI swelling and creep development (1 x 5 year) based upon real operational history—configuration of the fuel elements in the core, campaign—included in the RPV and RVI AMP. Measurement of the macroscopic geometry changes of the core shroud—under preparation, planned to the 2021–2022.

Country	Responses to Question 2
Japan	Failure events caused by radiation induced void swelling and creep have not been reported to NRA. A previous Japanese safety research program developed a creep evaluation formula based on creep tests of baffle former bolts under irradiation [7].
South Korea	We have yet no experiences or concerns of void swelling or creep in PWR components. Those might be a significant degradation mechanism in CANDU fuel channels.
Sweden	The baffle plates has been VT tested for void swelling and these test will be followed up every 4 year, no indications. Some material studies of Ringhals material has been done related to these issues, for example: Edwards et. al. J. of Nuclear Materials, vol 384, (2009), pp. 249–255. A. Jensen, B. Forssgren, P. Efsing, B. Bengtsson and M. Molin, "Examination of highly irradiated stainless steels from BWR and PWR reactor pressure vessel internals," Proceedings från Fontevraud 7, Avignon, Frankrike, SFEN, 2010.
Switzerland	Currently irradiation induced void swelling or creep is not an issue of the AMP and there is no running research programme. The phenomenon is mentioned in KATAM (catalogue of potential ageing mechanisms), but not addressed separately. ENSI will follow the state-of-the-art and the international OpEx. If necessary it will be included into the AMP.
U.S.A.	NRC and the U.S. industry have been interested in research activities related to both void swelling and creep. Void swelling of high-fluence baffle plates were assessed as part of the Zorita Internals Research Program (ZIRP). There has been no operating experience where void swelling has been indicated as a causal factor in the material/component degradation. While the NRC has not been actively involved in creep/stress relaxation research, the US industry, through EPRI, has been engaged in several recent projects. Stress relaxation does play a fundamental role in baffle-former-bolt failures. An extensive test matrix was developed to develop fundamental material stress relaxation data under representative temperature and irradiation conditions to support the improvement of baffle-former-bolt degradation models to inform both inspection and replacement activities. Aging management of void swelling and creep are managed by GALL Report (NUREG-1801, Rev. 2) AMP XI.M16A, PWR Vessel Internals, and GALL-SLR Report (NUREG-2191) AMP XI.M16A, PWR Vessel Internals.

Table 3-3 compiles responses from participating countries to Question 3 about radiation embrittlement and their (a) establishment and (b) application of embrittlement trend curves (ETCs).

Table 3-3. Embrittlement Trend Curves

Country		Responses to Question 3 (a, b)
Czech Republic	(a)	At this moment the current NTD ASI development is ongoing based on results from "IAEA NULIFE—Plant Life Management of NPPs" project, where the VERLIFE procedure was further modernised and extended with collaboration of experts of PWR operating countries as well as of VVER operating countries including Russian experts. The next edition of NTD ASI is prepared for issue in 2019. New prediction formulae of trend curves are based on results from the analysis of the database of surveillance specimen test results (after re-analysis of neutron fluence and reconstitution of specimens realized within European projects TACIS, TAREG and Russian projects).
	(b)	We use ETC for predictive purposes.

Country		Responses to Question 3 (a, b)
Japan	(a)	Current Japanese embrittlement trend curve is specified in the Japan Electric Association Code “Method of Surveillance Tests for Structural Materials of Nuclear Reactors (JEAC4201-2007) [2013 Supplement]” [8]. It was implemented into Japanese regulation in 2015.
	(b)	In Japan, ETC is used for interpolating between surveillance results when the fluence of the inner surface of RPV is greater than 2.4×10^{19} n/cm^2 (E > 1 MeV) [9].
South Korea	(a)	Embrittlement trend curve (ETC) in RG-1.99 (rev. 2) has been used as a Korean trend curve since RG-1.99 (rev. 2) was issued. ETC from 10 CFR 50.61a is only used for a comparative purpose.
	(b)	South Korea has used the embrittlement trend curve in RG-1.99 (rev. 2) to predict ΔRT_{NDT} and decrease in upper shelf energy at the design phase. While the plant operates, the trend curve of the specific reactor vessel can be adjusted by using the surveillance test results. In South Korea, all PWRs have their own surveillance specimens (capsules) as well as archive materials.
Sweden	(a)	It’s up to the licensee to justify the use of trend curve and implemented margin to the regulator. Because of RPV welds with high Ni- and Mn-content, Ringhals use an in-house ETC fitted to surveillance data.
	(b)	For weld material we rely on surveillance data and interpolating between results.
Switzerland	(a)	The applied ETC is the one defined by RG 1.99 Rev. 2. It was implemented in Swiss Regulation since 2008 (DETEC Ordinance “Preliminary shut down of NPP”) and 2011 (RegGuide ENSI-B01). The RG 1.99 ETC was used before 2008 too, but it was nonmandatory.
	(b)	If surveillance data are available (which is the case for all Swiss NPP), ETC is used only for interpolating/extrapolating these results.
U.S.A.	(a)	The methodology provided in Regulatory Guide 1.99, Rev. 2 (May 1988) is used to evaluate neutron embrittlement of RPV steels. Because use of a regulatory guide is not required, implementation is on a plant-specific basis, generally in plant-specific responses to Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations, issued on July 12, 1988. This guidance is often used to determine the allowable pressure-temperature envelopes for normal operations and BWR leak-testing that are required to satisfy Appendix G to 10 CFR Part 50—Fracture Toughness Requirements. Additionally, the procedures documented in RG 1.99, Rev. 2 are directly incorporated into 10 CFR 50.61—Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (1984), which requires licensees to demonstrate that the likelihood of reactor pressure vessel rupture during a pressurized thermal shock event is insignificant.
	(b)	The methodology of RG 1.99, Rev. 2, uses the results of surveillance testing (Regulatory Position C.2 of the RG), if the data are “credible” as defined in the RG, to best fit the ETC that is provided in Regulatory Position C.1 of the RG to the surveillance data. If the surveillance testing is not deemed credible, then the ETC is determined based solely on Regulatory Position C.1 using either known or generic information about the material. In either case, the ETC can be used to extrapolate the predicted shift in RT_{NDT} beyond the fluence values of the surveillance results.

Table 3-4 contains responses to Question 4 regarding plans to use additive manufacturing (or other advanced manufacturing techniques) for repair/replacement of components.

Table 3-4. Advanced Manufacturing Technologies

Country	Responses to Question 4
Czech Republic	Yes, validation of welding methodology for RPV wall repair was performed.
Japan	Answer from Central Research Institute of Electric Power Industry (CRIEPI): The Japanese utilities and CRIEPI have not developed repair and replacement technology that utilizes advanced technology, and have no plan for it.
South Korea	We plan to apply the additive manufacturing (AM) methods to the production of various reactor components, which include a nuclear fuel support grid, a lower flow plate, safety-grade valves and some parts of CRDM. National R&D programs are underway to evaluate the potential use of AM components in the reactor environments. [The South] Korean regulatory body is interested in advanced manufacturing techniques and thus, observing industry research results.
Sweden	A Vattenfall R&D project has been started together with the Swedish industry to manufacture some test specimens and investigate the material properties. No results have yet been published.
Switzerland	The ENSI as regulator does not plan or initiate the usage of such technologies. The initiative for the use of advanced manufacturing techniques has to come from the licensees and/or manufacturers. We currently have no knowledge of such plans. ENSI assumes these technologies offers attractive opportunities and that there will be interest in their future application. Therefore, ENSI will make an effort to keep up to date on advances in these technologies.
U.S.A.	The future use of advanced manufacturing techniques (AMT) in component repair and replacement in operating plants in the USA is currently uncertain. At least one component, a non-safety related thimble plug device (TPD), manufactured using laser powder bed fusion additive manufacturing, was installed at a plant in spring 2020. It is apparent that the U.S. nuclear industry is considering using AMTs for localized and surface applications such as localized repair, component hard-facing, and corrosion protection. Bulk AMT applications are also being considered for producing components that are no longer available from the original equipment manufacturer or that have excessive lead-times when manufactured using conventional methods.

Table 3-5 provides additional supporting references supplied by the participants from Japan for the responses summarized in Tables 4–7.

Table 3-5. Additional Japanese References

References Cited in Response to Questions 1, 2, 3, and 4
Reference (only in Japanese): [1] Nuclear Regulation Authority, The Guide for Extension of Operational Period on Commercial Power Reactors, September 20, 2017. https://www.nsr.go.jp/data/000069250.pdf [2] Nuclear Regulation Authority, The Standard Review Plan for Extension of Operational Period on Commercial Power Reactors, April 2016. https://www.nsr.go.jp/data/000147250.pdf [3] Nuclear Regulation Authority, The Guideline on Implementing Measures for Aging Management at Commercial Nuclear Reactors, September 20, 2017. https://www.nsr.go.jp/data/000069249.pdf [4] Nuclear Regulation Authority, The Standard Review Guideline for Aging Management at Commercial Nuclear Reactors, September 2, 2016. https://www.nsr.go.jp/data/000168877.pdf [5] Atomic Energy Society of Japan, Code on Implementation and Review of Nuclear Power Plant

References Cited in Response to Questions 1, 2, 3, and 4

Ageing Management Program 2015, AESJ-SC-P005:2015, March 2016.
[6] Nuclear Regulation Authority, Draft of field of safety research to be promoted in the future and its implementation policy, September 3, 2019.
https://www.nsr.go.jp/data/000275651.pdf
[7] Japan Nuclear Safety organization, Research report related to evaluation technique of irradiation assisted stress corrosion cracking (IASCC) JFY 2008, September 2009.
http://warp.da.ndl.go.jp/info:ndljp/pid/10207746/www.nsr.go.jp/archive/jnes/atom-pdf/seika/000014676.pdf
[8] The Japan Electric Association, Method of Surveillance Tests for Structural Materials of Nuclear Reactors (JEAC4201-2007) [2013 Supplement], May 14, 2014.
[9] Nuclear Regulation Authority, Technical evaluation report related to the Japan Electric Association's Method of Surveillance Tests for Structural Materials of Nuclear Reactors (JEAC4201-2007) [2013 Supplement], October 2015.
http://www.nsr.go.jp/data/000125554.pdf

4 WORKSHOP SUMMARY

The International Workshop on Age-Related Degradation of Reactor Vessels and Internals held May 23–24, 2019, at NRC Headquarters, focused on materials degradation of safety-related components during LTO, including RPV embrittlement and the degradation of RPV internals and piping due to irradiation. Participants included representatives from 31 regulatory, research, and industry organizations in 11 countries (Table 1). Presentations addressed the state of knowledge, research activities, and OE related to RPV embrittlement at high fluence levels, as well as degradation of RPV internals and other safety-significant primary pressure boundary components during operation for up to 80 years (Table 2). RES staff also worked with EPRI to access research insights and industry OE from its members, including both domestic and international NPP operators. Following the prepared presentations, the workshop participants were invited to participate in panel discussions addressing the prognosis for effective aging management and any additional research needs and to submit additional information in correspondence. Section 3 of this report includes these responses. The individual presentations from the workshop are summarized in Section 2, and slides are condensed in Appendix B. The participants’ presentation files are publicly available in ADAMS at Accession No. ML19150A174.

While the workshop presenters varied greatly among government, academic, and industry organizations, their experiences with materials aging management were remarkably similar. Each organization recognized several materials degradation challenges that would need to be addressed to enable LTO of power reactors, and they discussed technical and regulatory approaches to understanding and mitigating these challenges. The key technical issues identified as concerns for LTO were consistent across organizations and consistent with the high-priority topics being pursued by the NRC. Specifically, the main structures and components recognized as having materials challenges in extended operating lifetimes are the RPV, RVIs and piping, concrete, and cables.

Approaches to the development and use of ETCs for RPV embrittlement did vary somewhat. For example, while the United States requires licensees to demonstrate that the likelihood of RPV rupture during a pressurized thermal shock event is insignificant, technical guidance is in nonmandatory documents, and implementation is on a plant-specific basis. Some licensees use ETCs to interpolate between surveillance data points, while others also use ETCs for extrapolation beyond available data.

The workshop provided an opportunity to share information among international counterparts on approaches to aging management, including AMPs and related research being pursued by international organizations. These were found to be consistent with the NRC’s approach. The staff learned that international counterparts are evaluating similar technical issues, as their reactors enter periods of extended operation.

APPENDIX A WORKSHOP ATTENDEES

Table A-1 lists the workshop attendees, along with their organization and e-mail address.

Table A-1. Workshop Attendees

First (Given) Name	Last (Family) Name	Organization	E-Mail
David	Alley	NRC	david.alley@nrc.gov
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Roch	Menand	EDF	roch.menand@edf.fr

First (Given) Name	**Last (Family) Name**	**Organization**	**E-Mail**
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APPENDIX B PRESENTATION SLIDES

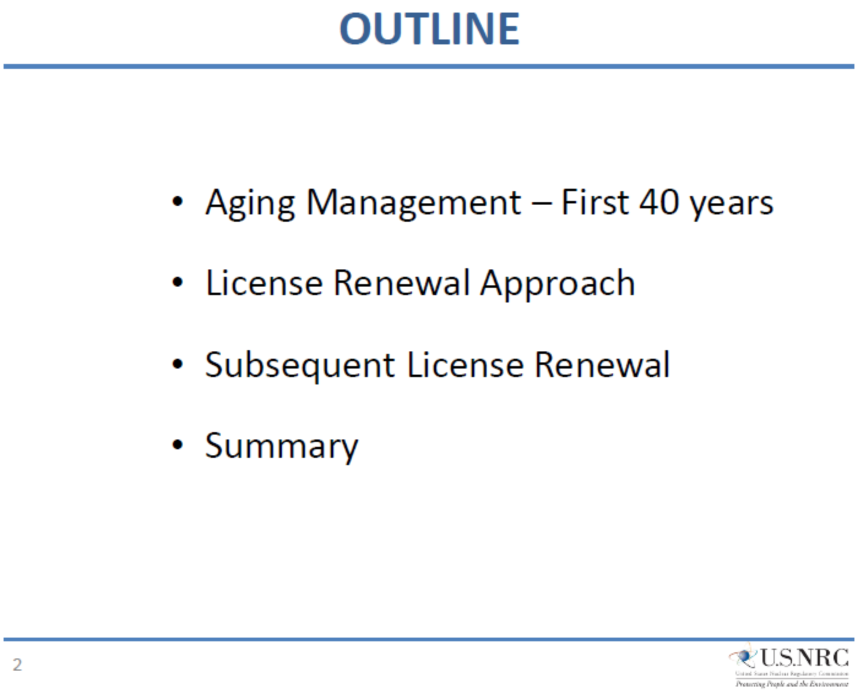
B.1 Aging Management and Subsequent License Renewal in the United States (A. Hiser)

Aging Management and Subsequent License Renewal in the United States

Allen Hiser, Jr.
Senior Technical Advisor

Division of Materials and License Renewal
Office of Nuclear Reactor Regulation

NRC International Workshop on Age-Related Degradation of Reactor Vessels and Internals
May 23, 2019



OUTLINE

- Aging Management – First 40 years
- License Renewal Approach
- Subsequent License Renewal
- Summary

WHAT IS AGING MANAGEMENT?

- Design, fabrication, construction, installation, testing and operation steps to:
  - Inhibit or preclude aging degradation (e.g., material selection and environmental control)
  - Identify degradation conditions in SSCs prior to a loss of intended function
  - Ensure effective corrective actions

HOW IS AGING MANAGED POST-START UP?

- First 40 years – Part 50
  - Regulation
  - NRC actions (orders, etc.)
  - Industry voluntary initiative
  - Emphasis on safety-related and RCS
- License Renewal – Part 54
  - Builds upon aging management activities implemented for first 40 years
  - Expands the scope of SSCs covered by aging management

FOR FIRST 40 YEARS (1/2)

- Requirements
  - 50.55a: ASME Code inspections
  - 50.65: Maintenance rule
  - 50.49 (EQ), 50.61 & 61(a) (PTS), Appendix G (fracture toughness requirements), Appendix H (RPV surveillance requirements) . . .
- NRC actions
  - Orders
    - Upper head CRDM nickel alloy penetrations
  - Generic Letter
    - GL-16-01, Monitoring of Neutron-Absorbing Materials in Spent Fuels Pools

FOR FIRST 40 YEARS (2/2)

- Industry voluntary initiatives
  - Reduces need for NRC to impose requirements
  - BWR Vessels and Internals Program
    - Vessel internals cracking, etc.
  - Materials Reliability Program
    - Baffle former bolts
  - Deviations subject to NRC notification

LICENSE RENEWAL RULE – 10 CFR PART 54

- A limited scope review based on “Regulatory Process Essential Elements”
- Rule provisions
  - Permits renewal for up to 20 years (e.g., 40 to 60 years)
  - Can apply 20 years before license expiration per 54.17(c)
  - Must apply at least 5 years before expiration per 2.109(b)
  - **A renewed license may be subsequently renewed per 54.31(d)**
  - No restrictions on number of subsequent renewals
- Focus is on demonstrating adequate management of the effects of aging for long-lived, passive structures and components important to plant safety
  - Other aspects of original license are not reconsidered
  - “A program based solely on detecting structure and component failures is not considered an effective aging management program”

LICENSE RENEWAL SAFETY PRINCIPLES

- Plant safety assured by regulatory process
  - Same plant operating rules for the renewal term
  - The plant’s current licensing basis (CLB) to be maintained
- Requires additional actions for aging management of passive, long-lived plant structures and components for license renewal

REGULATORY PROCESS ESSENTIAL ELEMENTS

- Effective compliance with regulations
- On-site resident inspectors and specialized inspections
- Performance assessments of inspection findings
- Operating experience analysis and utilization
- Safety issue resolutions (generic and plant specific)
- Materials aging & degradation issues important to safety addressed by
  - Rule changes, generic communications, orders, voluntary actions

BASIS FOR RULE

- Statement of Considerations – 1991 Rule
  - 56 Federal Register 64943; December 13, 1991
- NUREG-1412
  - "Foundation for the Adequacy of the Licensing Bases – A Supplement to the Statement of Considerations for the Rule on Nuclear Power Plant License Renewal (10 CFR Part 54)"
  - ADAMS Accession No. ML080310668
- Statement of Considerations – 1995 Rule
  - 60 Federal Register 22461; May 8, 1995

SCOPE OF LICENSE RENEWAL

- Safety-related systems, structures, and components (SSCs)
  - Maintain integrity of the reactor coolant pressure boundary
  - Ensure capability to shut down and maintain safe shutdown
  - Prevent or mitigate offsite exposures comparable to 10 CFR Part 100

- Non-safety related SSCs whose failure could affect safety-related SSC functions

- SSCs relied upon for compliance with the Commission's regulations for:
  - Fire Protection (10 CFR 50.48)
  - Environmental qualification (10 CFR 50.49)
  - Pressurized thermal shock (10 CFR 50.61)
  - Anticipated transients without scram (10 CFR 50.62)
  - Station blackout (10 CFR 50.63)

SCOPING & SCREENING

- Process for determining which structures and components require an aging management review (AMR)
  - Scoping – Identify SSCs that are within scope
  - Screening – Include only those structures and components (SCs) that are passive and long-lived
    - Passive = perform an intended function without moving parts or without a change in configuration or properties
    - Long-lived = not subject to replacement based on a qualified life or specified time period

- Reliability and performance of active structures and components are covered by compliance with the Maintenance Rule (10 CFR 50.65)

SAFETY REVIEW

- Review adequacy of scoping and screening
- Assess adequacy of aging management review – have all relevant aging effects been identified and will they be effectively managed (using aging management programs)
- Adequacy of evaluation of time-limited aging analyses (TLAAs)
- Inspections on licensee implementation of the license renewal aging management programs

LICENSE RENEWAL GUIDANCE DOCUMENTS

NUREG–2191, Vol. 1

Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report

Final Report

NUREG–2191, Vol. 2

Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report

Final Report

NUREG–2192

Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

Final Report

- These documents are subject to change using the License Renewal-Interim Staff Guidance (ISG-LR) process

GALL-SLR REPORT

- Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report
  - NUREG-2191, issued 2017
  - Provides assessments for aging management review, including identification of materials, environments and aging effects that require management
  - Identifies acceptable Aging Management Programs (AMPs)
  - Defines terms (structures and components, materials, environments, aging effects, significant aging mechanisms)

SRP-SLR

- Standard Review Plan for Subsequent License Renewal (SRP-SLR)
  - NUREG-2192, issued 2017
  - Guidance for NRC staff review of
    - Scoping and Screening
    - Aging Management Review
    - Time-limited Aging Analyses (TLAAs)
      - e.g., metal fatigue, reactor pressure vessel (RPV) neutron embrittlement, environmental qualification
    - UFSAR supplement description of AMPs
    - UFSAR supplement description of TLAAs

STANDARDS FOR APPROVAL

A renewed license may be issued if the Commission finds that:

- Actions have been identified and have been or will be taken such that there is **reasonable assurance** that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB. The actions are with respect to
  - managing the effects of aging during the period of extended operation on the functionality of structures and components
  - time-limited aging analyses
- Requirements for Environmental review have been satisfied
- Any consideration of Commission rules and regulations in adjudicatory proceedings has been resolved

SUBSEQUENT LICENSE RENEWAL (SLR)

- Operation from 60 to 80 years
- Regulatory framework and approval process is the same as license renewal
- Specific regulatory documents have been issued – GALL-SLR and SRP-SLR
- Certain issues up to plant-specific resolution
- Optimization of the review process
  - 18 month review (reduced from 22 months)
  - Increased use of in-office audits and web portals
  - Only final SER issued (previously included an “open items” SER)

CONCERNS FOR AGING

- Identification of potential new aging phenomena
  - Known mechanisms
  - New phenomena
- Approaches for identifying potential aging phenomena
  - Expanded materials degradation assessment
  - Results from 1st renewal aging management programs
  - Domestic and international operating experience
  - Research findings

NRC ACTIONS ON SUBSEQUENT RENEWAL

- Expert panel process to identify potential materials degradation issues for 80 years of operation
- Audits to assess results from implementation of AMPs at three plants with renewed licenses
- Public meetings with industry on technical issues, including operating experience and industry research activities
- NRC staff review of information and propose aging management approaches for 80 years of operation

BASIS FOR CHANGES TO DEVELOP GALL-SLR AMPS

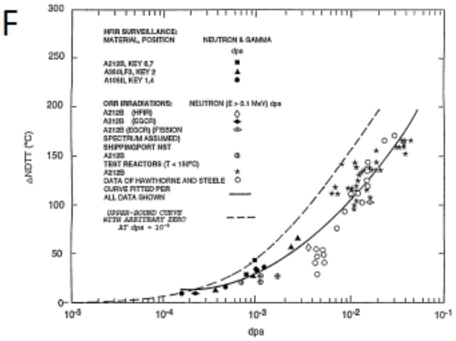
- To reflect expected aging differences for increased operating time from 60 to 80 years
- New plant operating experience since GALL Rev. 2
- Gaps identified in current guidance
- Improvements in efficiency and effectiveness of applications and NRC reviews
- Corrections to GALL Rev. 2 and SRP-LR Rev. 2
- Incorporate Interim Staff Guidance since GALL Rev. 2

SLR TECHNICAL ISSUES

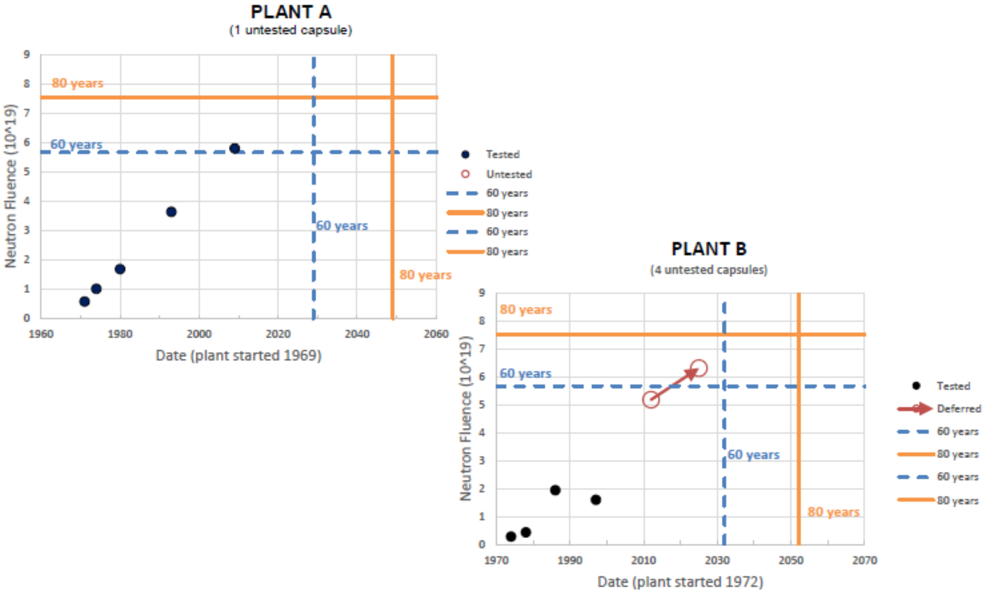
- **Reactor pressure vessel – neutron embrittlement**
  - Trends for high fluence levels
  - Surveillance programs
- **Reactor vessel internals - high fluence effects**
  - Irradiation-assisted stress corrosion cracking
  - Loss of fracture toughness
  - Void swelling
- Concrete and containment performance
  - Long-term radiation and high temperature exposure
  - Alkali-silica reaction (ASR)
- Electrical cables
  - Environmental qualification
  - In-service testing of cables
  - Long-term submersion of low and medium voltage cables

NEW TOPIC FROM SLR APPLICATIONS

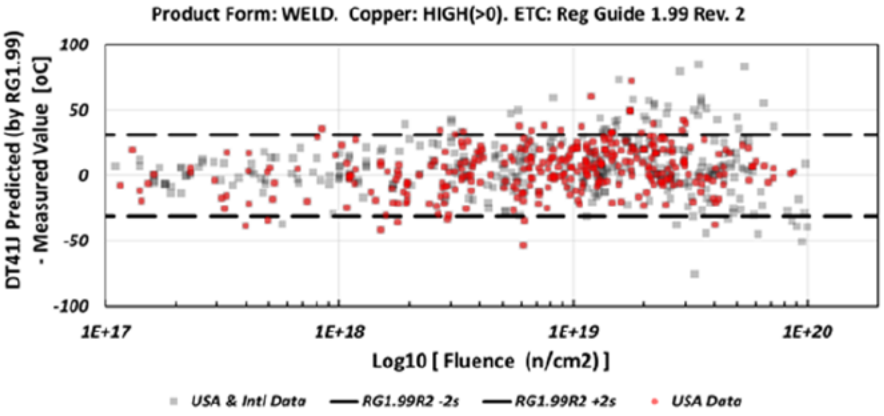
- Neutron embrittlement for reactor pressure vessel support steel elements
  - First SLR applicants have no concrete “shielding” between the RPV and the support steel
  - Low temperature – 150°F
  - Copper 0.30 wt-%
  - NUREG-1509 analysis



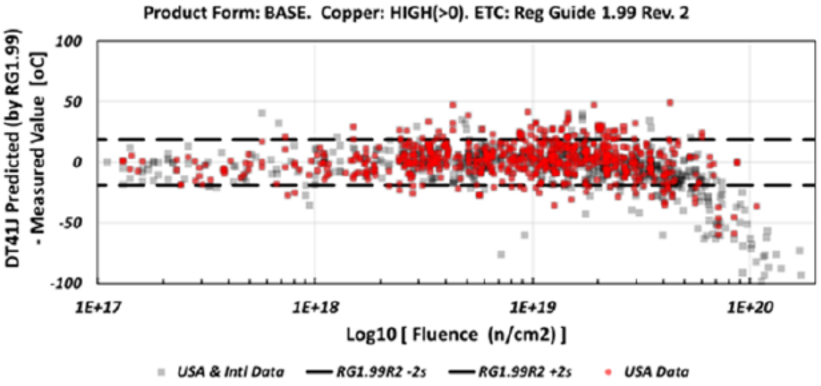
Reactor Vessel Material Surveillance AMP



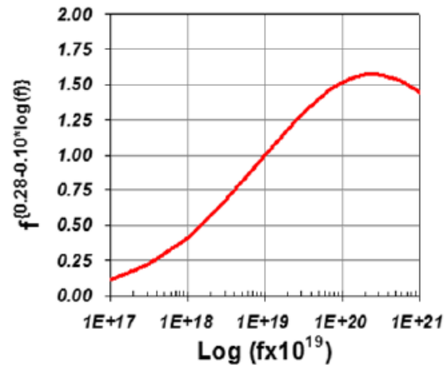
RESIDUALS FOR WELD MATERIALS



RESIDUALS FOR BASE MATERIALS



FLUENCE FACTOR RG 1.99, REV. 2



LICENSE RENEWAL STATUS

98 operating reactors

90 units with renewed licenses

49 units in the period of extended operation

STATUS OF SLR

- Three SLR applications under review
  - Turkey Point, Units 3 and 4 (PWR) – accepted May 2018
  - Peach Bottom, Units 2 and 3 (BWR) – accepted September 2018
  - Surry, Units 1 and 2 (PWR) – accepted December 2018
  - [North Anna, Units 1 and 2 (PWR) – expected 2020]
- Application subject to acceptance review:
  - To determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review.
  - To identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

SUMMARY

- License renewal is a limited scope review
  - Supported by Regulatory Process Essential Elements
- Licenses have been renewed for over 90% of U.S. plants and reviews of remaining plants (except two) are ongoing or planned
- Licensees are responsible for demonstrating that aging is adequately managed for licensed operating period
- NRC ensures that plants will be operated with reasonable assurance of adequate protection of the public health and safety
- NRC is reviewing SLR applications for 3 sites (6 units)

B.2 NRC's Aging and Materials Research Activities (R. Tregoning)

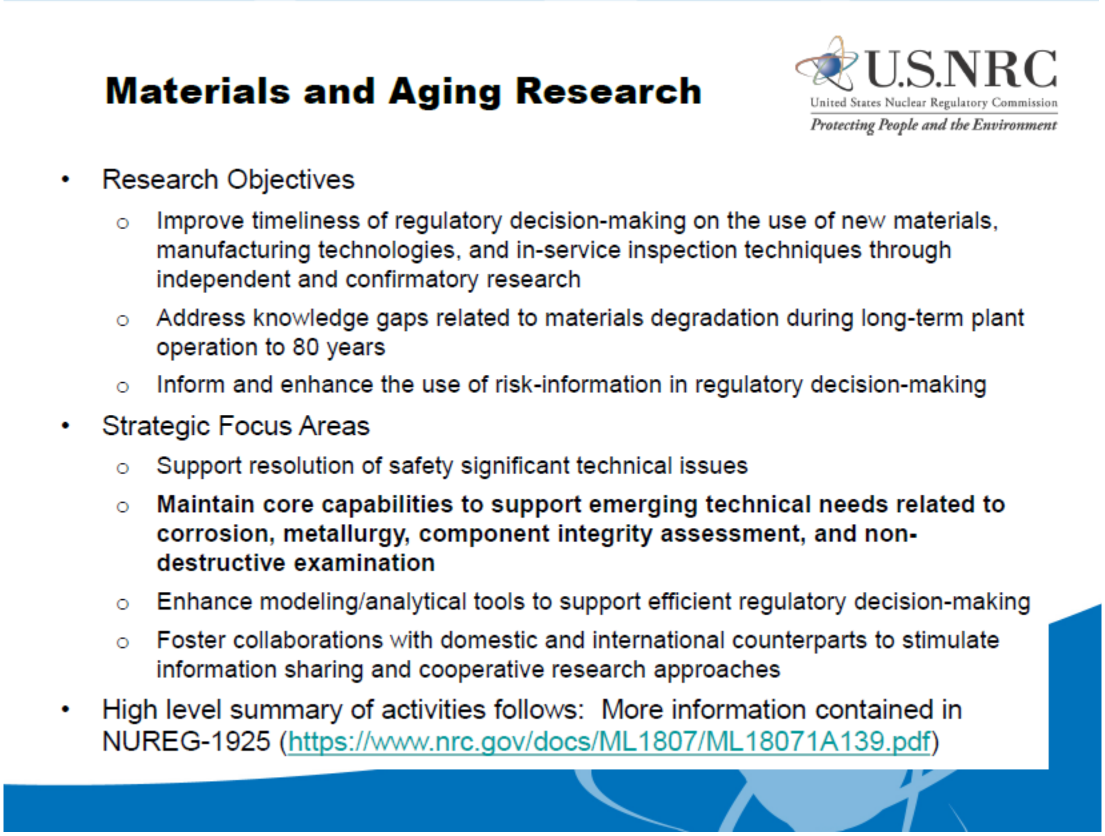
NRC's Aging and Materials Research Activities

Robert L. Tregoning

Nuclear Regulatory Commission

NRC International Workshop on Age-Related Degradation of Reactor Vessels and Internals
May 23 - 24, 2019
NRC Headquarters Rockville, MD

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment



Materials and Aging Research

U.S.NRC
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- Research Objectives
  - Improve timeliness of regulatory decision-making on the use of new materials, manufacturing technologies, and in-service inspection techniques through independent and confirmatory research
  - Address knowledge gaps related to materials degradation during long-term plant operation to 80 years
  - Inform and enhance the use of risk-information in regulatory decision-making
- Strategic Focus Areas
  - Support resolution of safety significant technical issues
  - **Maintain core capabilities to support emerging technical needs related to corrosion, metallurgy, component integrity assessment, and non-destructive examination**
  - Enhance modeling/analytical tools to support efficient regulatory decision-making
  - Foster collaborations with domestic and international counterparts to stimulate information sharing and cooperative research approaches
- High level summary of activities follows: More information contained in NUREG-1925 (https://www.nrc.gov/docs/ML1807/ML18071A139.pdf)

Materials and Aging Research

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Points of Contact
  - Raj Iyengar, Branch Chief, Component Integrity Branch
  - Istvan (Steve) Frankl, Branch Chief, Corrosion & Metallurgy Branch
  - Rob Tregoning, Senior Level Advisor
- Research Areas
  - Environmentally Assisted Degradation
    - Primary Water Stress Corrosion Cracking
    - Irradiated Assisted Stress Corrosion Cracking
  - Steam Generator Tube Integrity
  - Aging Management during Long Term Operation
  - Spent Fuel Dry Storage
  - Neutron Absorber Materials in Spent Fuel Pools
  - Advanced Manufacturing Technologies for Reactor Components
  - Reactor Pressure Vessel Integrity
  - Piping Integrity
  - Probabilistic Component Integrity
  - Nondestructive Examination
  - *Advanced Non-light Water Reactors (ANLWR) Materials*

Primary Water Stress Corrosion Cracking (PWSCC)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

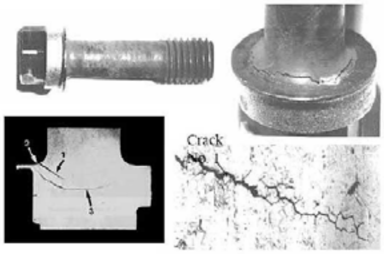
- Overview
  - **Objective:** Evaluate PWSCC crack initiation and crack growth rate (CGR) susceptibility of nickel-based alloys
  - **Motivation:** Provide assurance of reactor coolant pressure boundary integrity
  - **Regulatory Application:** Support reviews of proposed changes to the inspection requirements in the ASME Code and associated rulemaking
  - **Collaboration:** EPRI

- Recent Accomplishments
  - "Experimental Plan for Primary Water Stress Corrosion Crack Initiation Testing" (ADAMS ML15272A300)
  - NUREG/CR-7226, "Primary Water Stress Corrosion Cracking of High-Chromium, Nickel-Base Welds Near Dissimilar Metal Weld Interfaces" (ML18018A562)
- Next Steps
  - Publish initial report on Alloy 600/182 PWSCC initiation testing (2019)
  - Complete CGR testing of Alloy 690/52/152 (2020)
  - Publish NUREG/CR reports on CGR testing (2020)
  - Complete crack initiation testing of Alloys 600/182 and 690/52/152 (2021 – 2022)

Irradiation-Assisted Stress Corrosion Cracking (IASCC)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Overview
  - **Objective:** Evaluate IASCC degradation mechanisms during long-term operations (LTO)
  - **Motivation:** Confirm adequacy of reactor internal aging management programs
  - **Regulatory Application:** Support reviews of internals inspection/evaluation guidance, ASME Code changes and associated rulemaking
  - **Collaboration:** EPRI, Halden Reactor Project, International Regulators



- Recent Accomplishments
  - NUREG/CR-6909 Rev. 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials" (ML16319A004)
  - NUREG/CR-7185, "Effect of Thermal Aging and Neutron Irradiation on Crack Growth Rate and Fracture Toughness of Cast Stainless Steels and Austenitic Stainless Steel Welds" (ML15202A007)
- Next Steps
  - Complete cooperative research on Zorita reactor internal materials (2019-2020)
  - Complete initial testing of Zorita materials at ANL (2019 – 2020)
  - Identify opportunities for further irradiation or harvesting of reactor internals (2020)

Steam Generator Tube Integrity Program (SG-TIP)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Overview
  - **Objective:** Assess NDE reliability and associated tube integrity for emerging inspection procedures and plans
  - **Motivation:** Confirm adequacy of industry practices used for in-service inspection
  - **Regulatory Application:** Review acceptability of advanced techniques or implementation plans proposed by industry
  - **Collaboration:** EPRI, CNSC, KINS, KAERI, and IRSN
- Recent Accomplishments
  - NUREG/CR-7217, "Application of Automated Analysis Software to Eddy Current Inspection Data from Steam Generator Tube Bundle Mock-up" (ML16271A090)
  - NUREG/CR-7225, "Stability of Circumferential Flaws in Once-Through Steam Generator Tubes Under Thermal Loading During LOCA, MSLB and FWLB" (ML17324B296)
- Next Steps
  - Complete report summarizing eddy current inspections and pressure testing of U-bend tubes with PWSCC flaws (2019)
  - Complete report on detection of cracking near volumetric indications (2019)
  - Evaluate issues related to eddy current detection and auto-analysis techniques

Long-Term Operation (LTO) & Aging Management

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Overview
  - **Objective:** Support guidance development, coordinate related research activities, develop a systematic approach for harvesting materials and components from reactors
  - **Motivation:** Provide assurance that aging effects will be adequately managed during LTO
  - **Regulatory Application:** Refine, as appropriate, existing aging management plans and guidance
  - **Collaboration:** DOE and EPRI
- Recent Accomplishments
  - NUREG-2221, "Technical Bases for Changes in the Subsequent License Renewal Guidance Documents NUREG–2191 and NUREG–2192" (ML17362A126)
  - PNNL-27120 Rev. 1, "Criteria and Planning Guidance for Ex-Plant Harvesting to Support Subsequent License Renewal" (ML19081A006)
- Next Steps
  - Drafting joint roadmap for metals research (RPV, internals, piping) (2019)
  - Considering international workshop to coordinate harvesting activities (2019)
  - Planning international workshops on cables and concrete (2020 – 2021)

Spent Fuel Dry Storage

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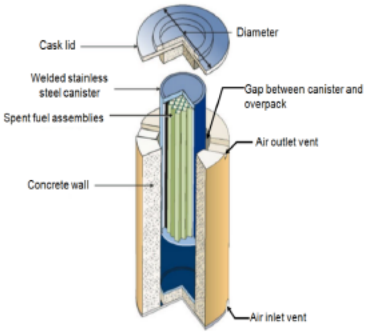
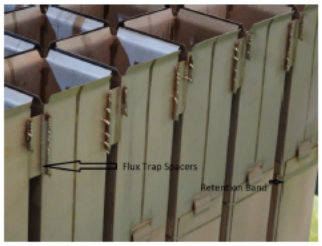
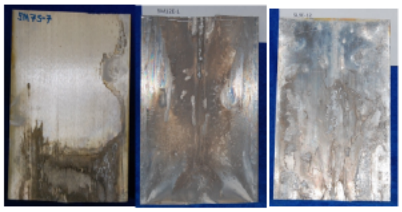


- Overview
  - **Objective:** Assess chloride-induced SCC (CISCC) and adequacy of associated NDE techniques
  - **Motivation:** Provide assurance of dry cask storage systems (DCSSs) integrity during extended storage
  - **Regulatory Application:** Support development of regulatory guidance for aging management of spent fuel DCSSs
  - **Collaboration:** DOE and EPRI
- Recent Accomplishments
  - NUREG/CR-7170, "Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts" (ML14051A417)
  - PNNL-24412 Rev. 1, "Nondestructive Examination Guidance for Dry Storage Casks" (ML16270A535)
- Next Steps
  - Finalize crack growth test plan for CISCC CGR testing planned in cooperation with DOE (2019)
  - Evaluate detection and sizing capability of in-situ NDE methods
  - Support evaluation of implementation of ISI requirements within ASME Code

Neutron Absorber Materials (NAM)



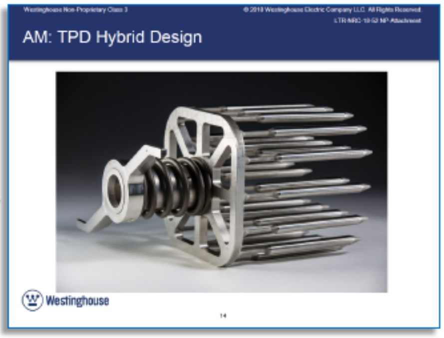
- Overview
  - **Objective:** Assess degradation mechanisms and surveillance methods for NAMs utilized for maintaining sub-criticality margins in spent fuel pools
  - **Motivation:** Support technical bases for regulatory decisions related to subsequent license renewal (SLR)
  - **Regulatory Application:** Support review of criticality safety analyses for SLR applications
  - **Collaboration:** EPRI
- Recent Accomplishments
  - Completed the evaluation of Boral® NAM panels obtained from the decommissioned Zion SFP.
  - SRNL-TR-2018-00244, "Characterization and Analysis of Boral® from the Zion Nuclear Power Plant Spent Fuel Pool" (ML19140A365 – available in July 2019)



- Next Steps
  - Release SRNL report on the evaluation of the Zion NAM panels (2019)
  - Publish report on the estimated total measurement uncertainty of ^{10}B areal density measurements using the BADGER system (2019)

Advanced Manufacturing Technologies (AMT)

- Overview
  - **Objective:** Develop sufficient expertise and guidance for efficient review of advanced manufacturing technologies for use in existing and future nuclear reactors
  - **Motivation:** Prepare NRC for expected submittals to use AMTs in nuclear applications
  - **Regulatory Application:** Support review of AMT submittals and development of associated guidance
  - **Collaboration:** DOE, EPRI, and NIST


Westinghouse Non-Proprietary Class 3

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LTR-NRC-18-52 NP-Attachment

AM: TPD Hybrid Design

- Recent Accomplishments
  - NUREG/CP-0310, "Proceedings of the Public Workshop On Additive Manufacturing For Reactor Materials And Components" (ML18221A109)
  - Published initial AMT Agency Action Plan (ML19029B355)
- Next Steps
  - Complete technical and regulatory documents for licensing candidate AMT (2019)
  - Develop knowledge management plan (2019)
  - Develop generic technical information needed in AMT submittals (2019)

Reactor Pressure Vessel (RPV) Integrity

- Overview
  - **Objective:** Continue development and verification of FAVOR and assess embrittlement prediction formulas
  - **Motivation:** Confirm continued integrity of RPV during LTO
  - **Regulatory Application:** Enhance guidance for RPV structural integrity and fluence calculations
  - **Collaboration:** JAEA, JNRA, and CSNI
- Recent Accomplishments
  - NUREG-2163, "Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule, Final Report" (ML15058A677)
  - RG-1.230, "Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule" (ML15344A402)
- Next Steps
  - Evaluate continued efficacy of Regulating Guide - 1.99, Revision 2 (2019)
  - Document fluence calculation methodology study for extended beltline and reactor internals (2019)
  - Complete FAVOR verification and validation and software quality assurance (2020)
  - Complete shallow flaw structural integrity investigation (2020 - 2021)

Piping Integrity

- Overview
  - **Objective:** Develop and enhance analytical methods and tools to assess structural integrity of reactor piping systems
  - **Motivation:** Confirm continued integrity of safety-critical piping systems during LTO
  - **Regulatory Application:** Enhance guidance for performing piping structural integrity calculations
  - **Collaboration:** EPRI and CSNI
- Recent Accomplishments
  - PVP2017-66104, "Finite Element Analysis of the Effect of Mechanical Stress Improvement Process on Weld Residual Stress and Flaw Growth in a Thick-Walled Pressurizer Safety Nozzle"
  - PVP2018-84931, "Exploring Finite Element Validation for Weld Residual Stress Prediction"
- Next Steps
  - Publish report on modeling guidance for weld residual stress modeling (2019)
  - Participate in CSNI benchmark projects on LBB, xFEM, and leak rate (2019 - 2020)
  - Implement research plan for validating xFEM-A for PWSCC predictions (2020)
  - Complete testing/evaluation of carbon fiber reinforced polymer repair methods (2020)

Probabilistic Structural Integrity (PSI)

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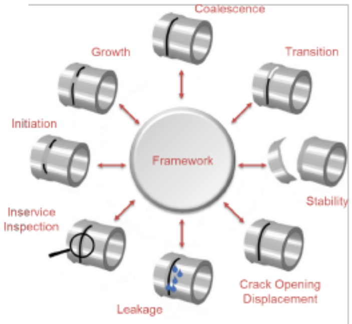

- Overview
  - **Objective:** Develop probabilistic methods to assess structural integrity of primary coolant pressure boundary (PCPB) components
  - **Motivation:** Confirm continued integrity of safety-critical piping systems during LTO
  - **Regulatory Application:** Develop guidance for performing probabilistic structural integrity analyses
  - **Collaboration:** EPRI, JNRA, JAEA, SSM, CNSC
- Recent Accomplishments
  - Technical Letter Report, "Important Aspects of Probabilistic Fracture Mechanics Analyses" (ML18178A431)
  - "Acceptance Criteria for Use with xLPR Version 2 Code" (ML16271A436)
  - "xLPR External Review Board Summary Documents & Close-out Letter" (ML17276A650)
- Next Steps
  - Complete efforts to publicly release xLPR (piping code) (2019)
  - Support regulatory acceptance review for xLPR (2019 – 2020)
  - Complete draft PSI regulatory guide with supporting technical basis report (2020)
  - Complete studies on leak-before-break applications with xLPR (2020 – 2021)

Non-Destructive Evaluation (NDE)

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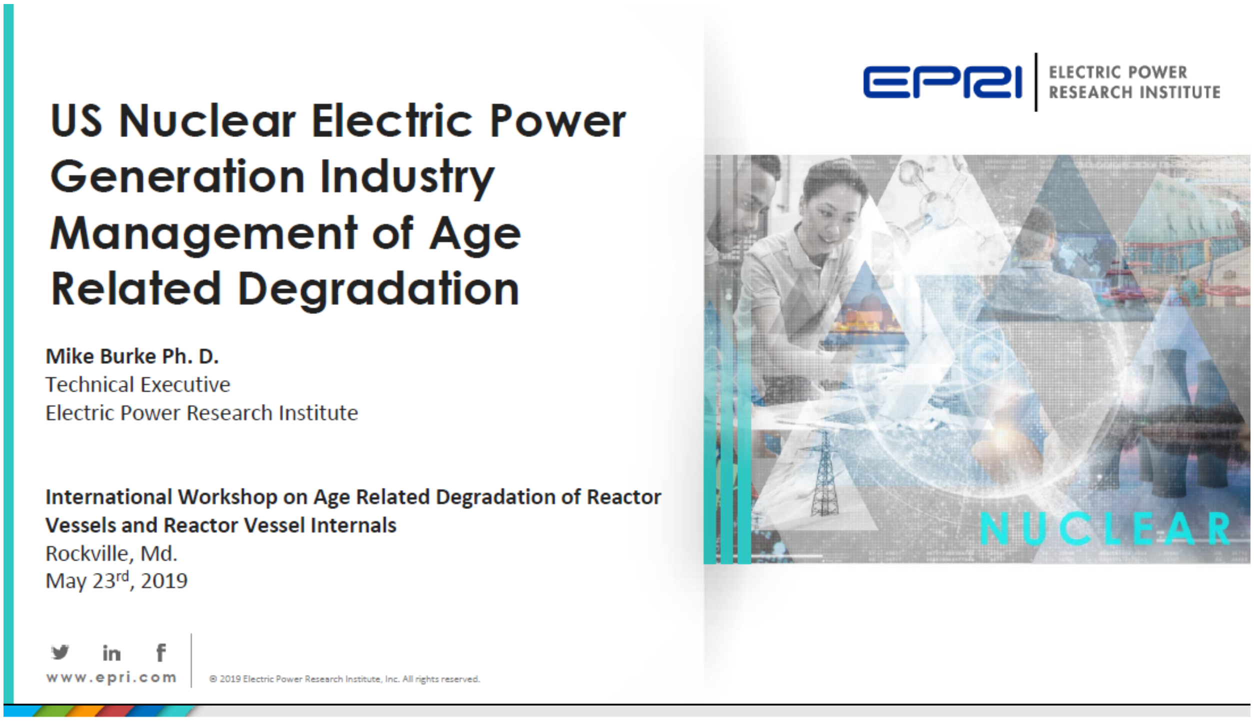
- Overview
  - **Objective:** Evaluate effectiveness and reliability of NDE techniques
  - **Motivation:** Confirm adequacy of industry procedures and practices
  - **Regulatory Application:** Support reviews of ASME Code modifications and proposed revisions of current requirements
  - **Collaboration:** EPRI, IRSN, and PIONIC
- Recent Accomplishments
  - PNNL-27441, "Human Factors in Nondestructive Examination: Manual Ultrasonic Testing Task Analysis and Field Research" (ML18176A055)
  - PNNL-27712, "Interim Analysis of the EPRI CASS Round Robin Study" (ML18219B319)
  - PNNL-28362, "Ultrasound Modeling and Simulation Status Update" (ML19010A072)
  - NUREG/CR-7235, "Results of Blind Testing for the Program to Assess the Reliability of Emerging Nondestructive Techniques" (ML17159A466)
- Next Steps
  - Complete state-of-the art report on inspecting CASS and welds in CASS (2019)
  - Conduct PIONIC – international follow-on to PARENT program (2019 – 2021)
  - Develop technical basis to support guidance for UT Modeling & Simulation (2022)

B.3 U.S. Nuclear Electric Power Generation Industry Management of Age-Related Degradation (M. Burke)



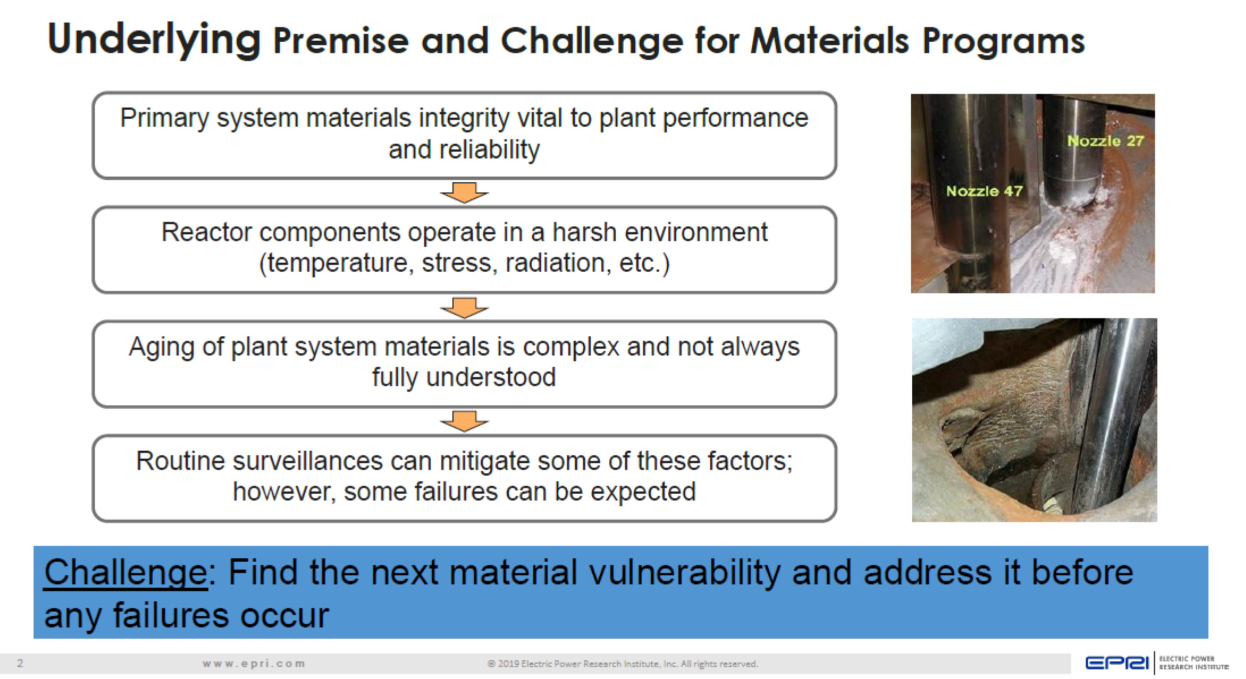
ELECTRIC POWER RESEARCH INSTITUTE

US Nuclear Electric Power Generation Industry Management of Age Related Degradation

Mike Burke Ph. D.
Technical Executive
Electric Power Research Institute

International Workshop on Age Related Degradation of Reactor Vessels and Reactor Vessel Internals
Rockville, Md.
May 23rd, 2019

www.epri.com



Underlying Premise and Challenge for Materials Programs

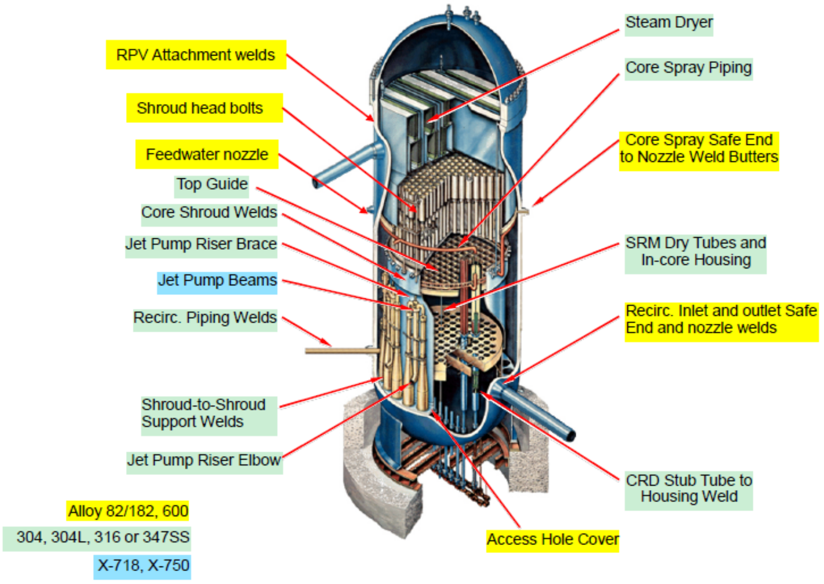
- Primary system materials integrity vital to plant performance and reliability
- Reactor components operate in a harsh environment (temperature, stress, radiation, etc.)
- Aging of plant system materials is complex and not always fully understood
- Routine surveillances can mitigate some of these factors; however, some failures can be expected

Challenge: Find the next material vulnerability and address it before any failures occur

Operating Experience through 2002

- Reactor Internals components:
  - Core shrouds
  - Core spray piping
  - Jet pump beams
  - Jet pump inlet piping
  - Top guides
  - Steam dryers
  - BWR fuel support castings
  - PWR baffle bolts and split pins
- Pressure boundary items:
  - BWR recirculation piping
  - BWR CRD stub tubes
  - Pressurizer heater sleeves
  - Primary system full penetration butt welds
  - PWR reactor vessel bottom mounted nozzles
  - PWR CRDM head penetrations

BWR IGSCC



RPV Attachment welds

Shroud head bolts

Feedwater nozzle

Top Guide

Core Shroud Welds

Jet Pump Riser Brace

Jet Pump Beams

Recirc. Piping Welds

Shroud-to-Shroud Support Welds

Jet Pump Riser Elbow

Steam Dryer

Core Spray Piping

Core Spray Safe End to Nozzle Weld Butters

SRM Dry Tubes and In-core Housing

Recirc. Inlet and outlet Safe End and nozzle welds

CRD Stub Tube to Housing Weld

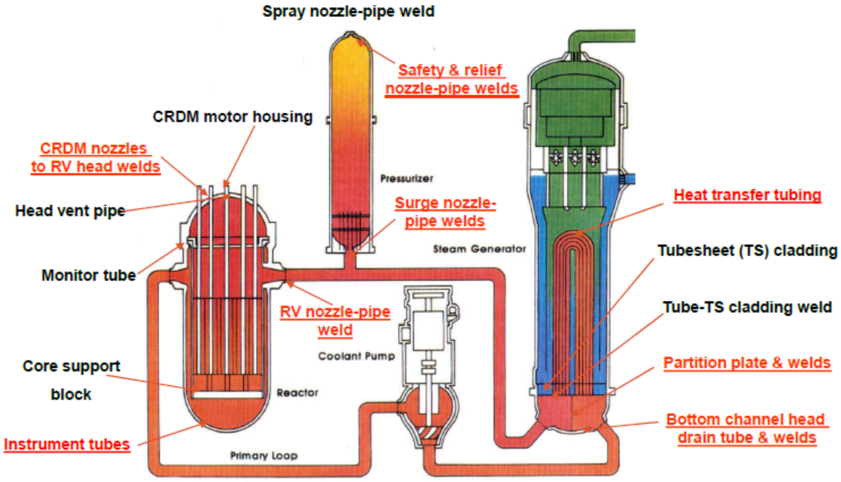
Access Hole Cover

Alloy 82/182, 600

304, 304L, 316 or 347SS

X-718, X-750

PWR PWSCC



Locations that have experienced cracking and or leakage are highlighted in red.

Call to Action – Early 2000's

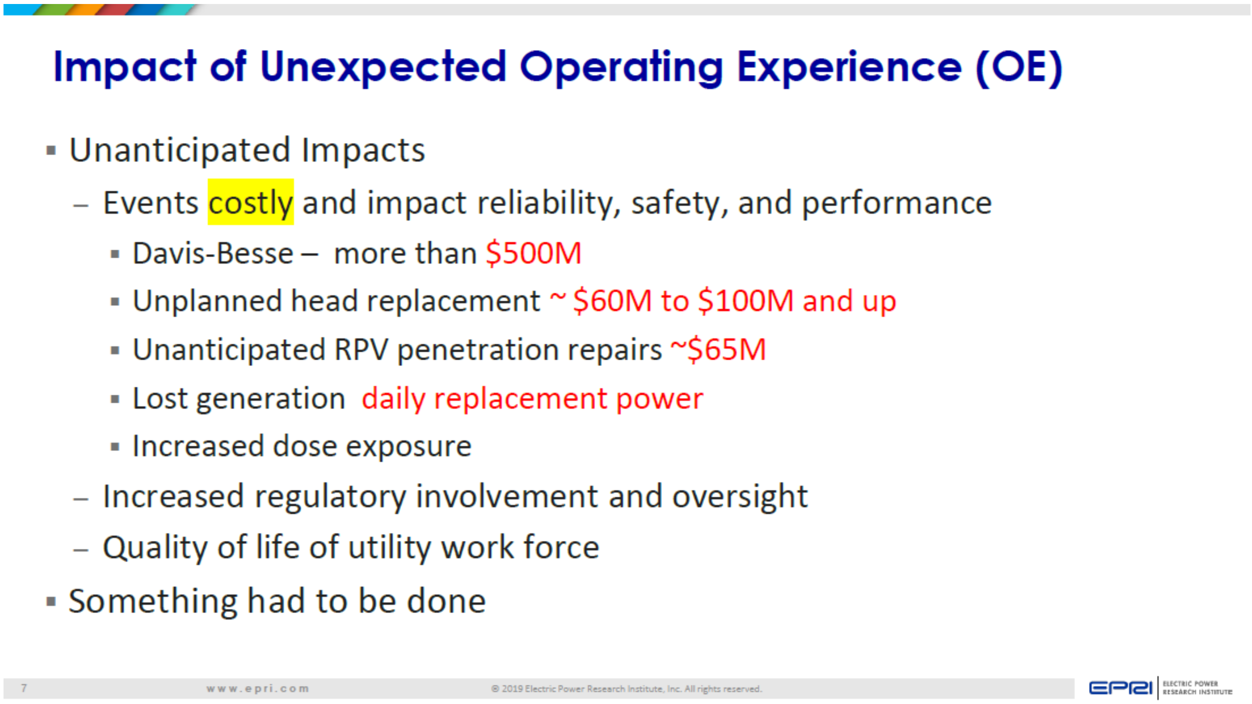
Series of Events in PWRs Motivated Industry Executives to Take Broad Generic Action

Indian Point (February 2000):
SG tube leak hidden by noise in NDE signal

V.C. Summer (Fall 2000):
Crack in a hot leg nozzle to pipe weld

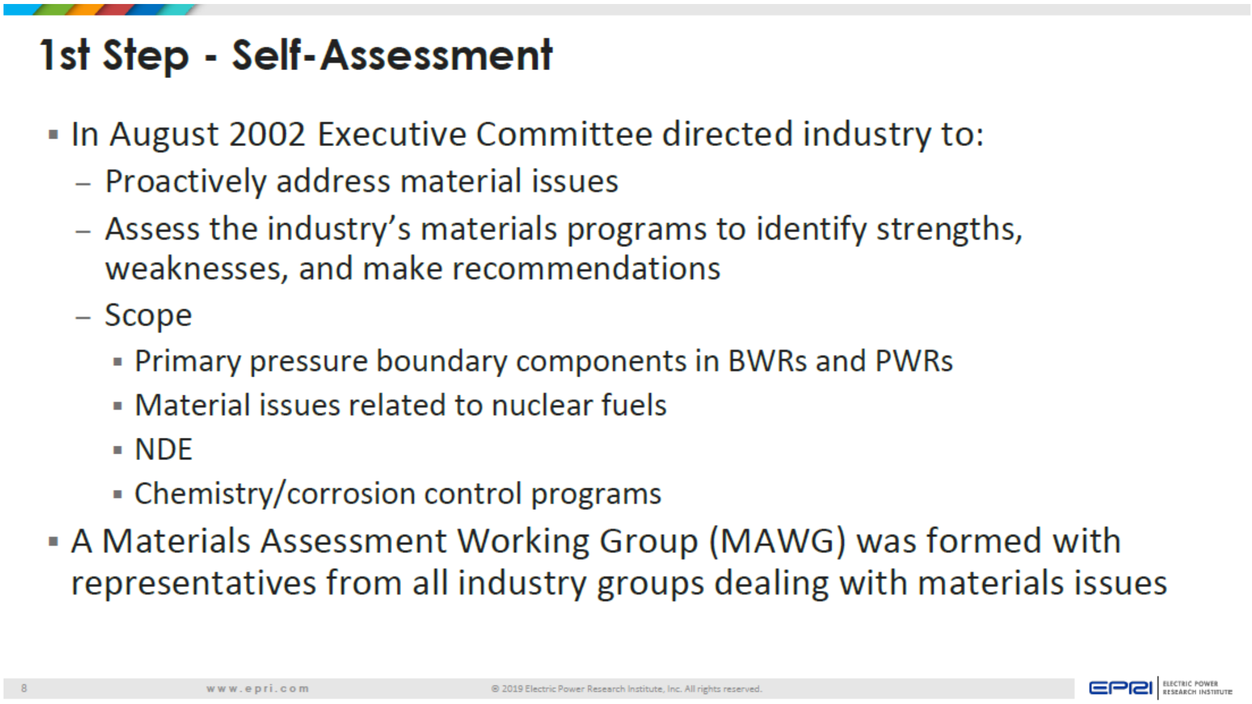
Leaking PWR CRDM Head Penetrations

Davis Besse (March 2002):
CRDM penetration and head wastage



Impact of Unexpected Operating Experience (OE)

- Unanticipated Impacts
  - Events costly and impact reliability, safety, and performance
    - Davis-Besse – more than $500M
    - Unplanned head replacement ~ $60M to $100M and up
    - Unanticipated RPV penetration repairs ~$65M
    - Lost generation daily replacement power
    - Increased dose exposure
  - Increased regulatory involvement and oversight
  - Quality of life of utility work force
- Something had to be done



1st Step - Self-Assessment

- In August 2002 Executive Committee directed industry to:
  - Proactively address material issues
  - Assess the industry's materials programs to identify strengths, weaknesses, and make recommendations
  - Scope
    - Primary pressure boundary components in BWRs and PWRs
    - Material issues related to nuclear fuels
    - NDE
    - Chemistry/corrosion control programs
- A Materials Assessment Working Group (MAWG) was formed with representatives from all industry groups dealing with materials issues

MAWG Issue Statement

The corrosion of the base metal in the Davis-Besse reactor pressure vessel head, increased occurrences of control rod drive mechanism nozzle cracking in pressurized water reactors (PWRs), and the V. C. Summer hot leg dissimilar metal weld defect represent issues that have threatened nuclear plant asset value and raised questions regarding the ability of the industry to detect degraded conditions in reactor coolant system components and piping. Other plant events over the last three years involving steam generator tubes, boiling water reactor (BWR) vessel internals and other pressure boundary, vessel internals, and fuel materials suggest that the nuclear industry has not been able to consistently anticipate and manage materials problems as well as it could. These events also suggest the need for better integration of existing PWR and BWR materials programs, as well as underlying technical support programs in the areas of plant chemistry, NDE, cracking/corrosion research, etc. Lessons from programs that are working well need to be transferred. Finally, the industry must continue to monitor and manage the impacts of materials issues prevention and mitigation strategies on fuel reliability and performance.

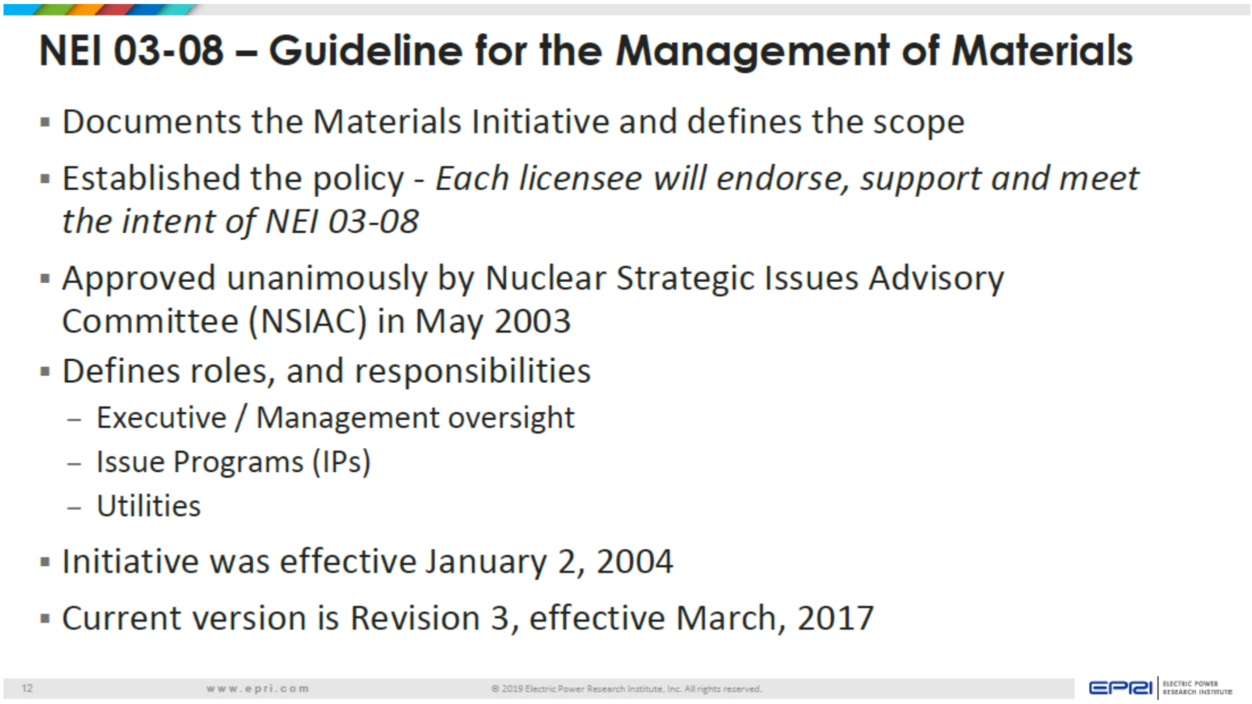
Key Conclusions from Self Assessment

- Industry lacked a unified strategic focus and direction
- Limited coordination of industry efforts on materials issues
- Budget and funding challenges
- Unable to enforce and to verify implementation of industry guidance
- Oversight of industry materials efforts was inconsistent
- Industry participation in materials issue programs lacking
- Implementation of materials tools inconsistent

- Recommendation: Define an NSIAC Initiative to address materials aging management

What is an NSIAC Initiative?

- Formal agreement among the utility Chief Nuclear Officers (CNOs) that form the Nuclear Strategic Issues Advisory Committee (NSIAC) to follow a defined policy
- Requires 80% vote of the NSIAC for approval
- <u>Binding</u> industry commitment at CNO level for full implementation



NEI 03-08 – Guideline for the Management of Materials

- Documents the Materials Initiative and defines the scope
- Established the policy - *Each licensee will endorse, support and meet the intent of NEI 03-08*
- Approved unanimously by Nuclear Strategic Issues Advisory Committee (NSIAC) in May 2003
- Defines roles, and responsibilities
  - Executive / Management oversight
  - Issue Programs (IPs)
  - Utilities
- Initiative was effective January 2, 2004
- Current version is Revision 3, effective March, 2017

Materials Management Policy Statement

"… the industry will ensure that its management of materials degradation and aging is ***forward-looking and coordinated*** to the maximum extent practical. Additionally, the industry will ***continue to*** rapidly identify, react and ***effectively respond to emerging issues***. The associated work will be managed to emphasize safety and operational risk significance as the first priority, appropriately balancing long term aging management and cost as additional considerations. To that end, as issues are identified and as work is planned, the groups involved in funding, managing and providing program oversight will ensure that the ***safety and operational risk significance of each issue is fully established prior to final disposition***."

Materials Initiative

- The objective is to assure safe, reliable and efficient operation of the U.S. nuclear power plants in the management of materials issues.
- The purpose of this Initiative is to:
  - Provide a consistent management process
  - Provide for prioritization of materials issues
  - Provide for proactive approaches
  - Provide for integrated and coordinated approaches to materials issues
- Utility actions required by this Initiative include:
  - Commitment of executive leadership and technical personnel
  - Commitment of funds for materials issues within the scope of this Initiative
  - Commitment to implement applicable guidance documents
  - Provide for oversight of implementation

NEI 03-08 Scope and Issue Programs

- Scope
  - Reactor internals
  - Primary system pressure boundary components
  - Related NDE, chemistry and corrosion controls
  - Other as directed by NSIAC
- Issue Programs (IPs)
  - EPRI
    - BWR Vessel & Internals Project (BWRVIP)
    - Materials Reliability Program (MRP)
    - Steam Generator Management Program (SGMP)
    - Nondestructive Evaluation (NDE)
    - Primary Systems Corrosion Research (PSCR)
    - Water Chemistry Control (WCC)
  - PWR Owners Group Materials Subcommittee

NEI 03-08 Expectations for Owners

- **Utility responsibilities** (shall)
  - Maintain a RCS Materials Degradation Management Program
  - Implement “Mandatory” and “Needed” IP guidance (see next slide)
  - Participate in IPs
  - Apply appropriate focus on materials issues
  - Communicate materials OE

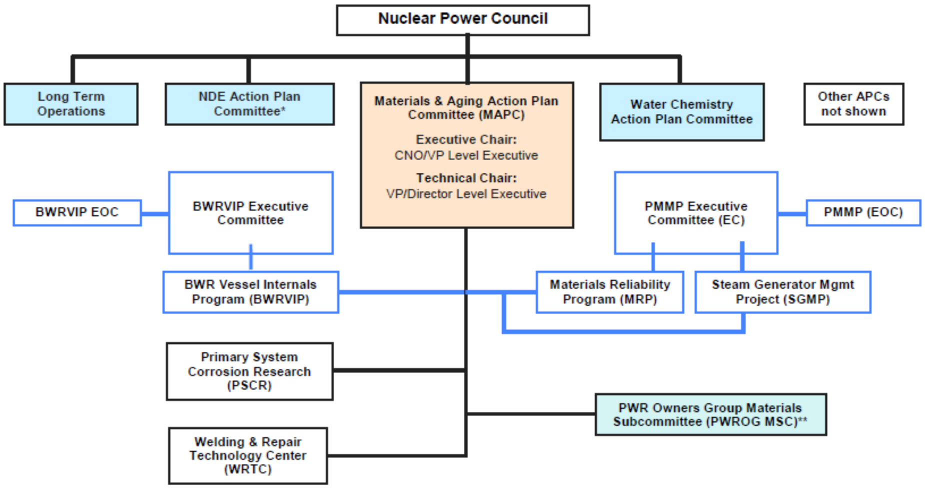
NEI 03-08 Expectations for Materials Programs

- Identifying, prioritizing, and resolving issues
- Communicating
- Managing regulatory interface
- Developing guidance
- Reviewing deviations
- Self assessments and performance metrics
- Process for addressing emergent materials issues

MAPC Structure

- EPRI Materials Action Plan Committee (MAPC) provides strategic direction for materials programs including:
  - EPRI materials programs: BWRVIP, MRP, SGMP, PSCR, WCC, NDE
  - PWROG MSC (Materials Committee)
- BWRVIP and PWR executive committees report to MAPC for strategic coordination
- MAPC members to include:
  - CNO as chair to coordinate with NSIAC
  - Executive and Technical chairs of the Materials Programs
  - At-large members for fleet representation
  - INPO and NEI representatives
- Effective January 1, 2010

Materials Organizational Structure



Nuclear Power Council

Long Term Operations

NDE Action Plan Committee*

Materials & Aging Action Plan Committee (MAPC)
Executive Chair: CNO/VP Level Executive
Technical Chair: VP/Director Level Executive

Water Chemistry Action Plan Committee

Other APCs not shown

BWRVIP EOC

BWRVIP Executive Committee

PMMP Executive Committee (EC)

PMMP (EOC)

BWR Vessel Internals Program (BWRVIP)

Materials Reliability Program (MRP)

Steam Generator Mgmt Project (SGMP)

Primary System Corrosion Research (PSCR)

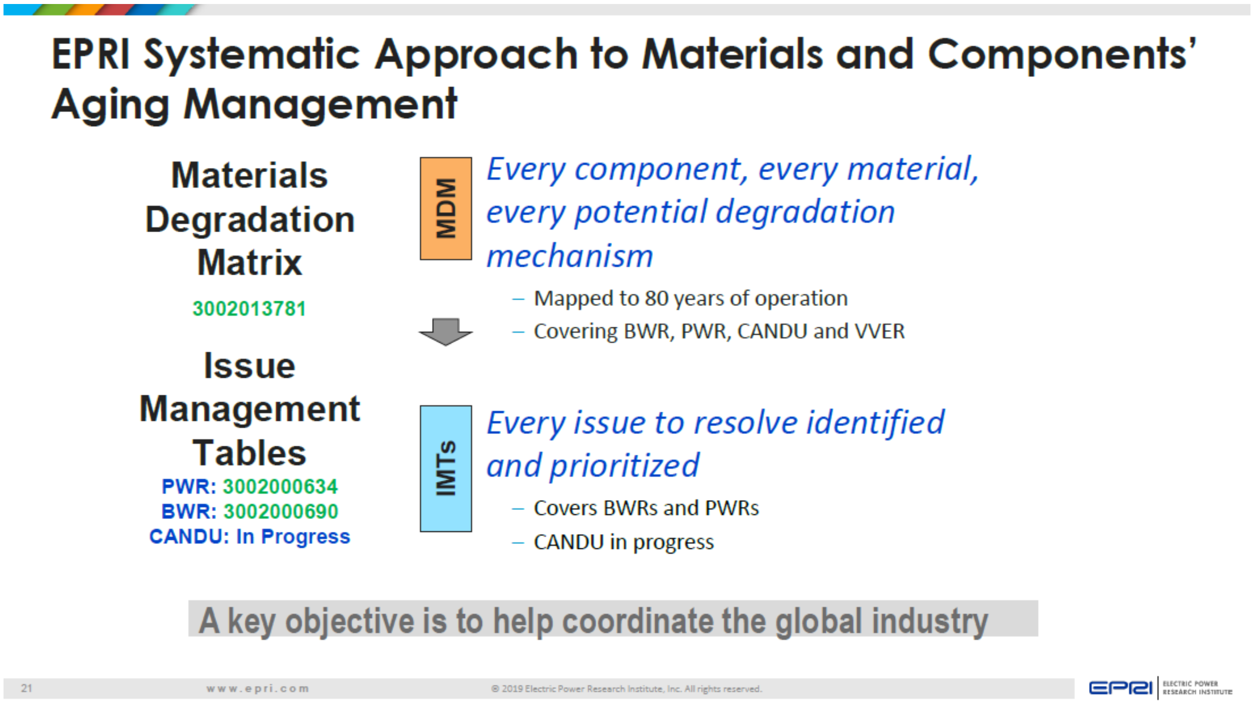
PWR Owners Group Materials Subcommittee (PWROG MSC)**

Welding & Repair Technology Center (WRTC)

*NDE APC coordinates with Materials APC and PWR Owners Group
**Materials Subcommittee has a representative on Materials APC

NEI 03-08 Integrated Materials Issues Strategic Plan

- A Systematic Approach to Managing Materials Issues
  - Identify degradation modes (DM)
  - Assessment of the extent of degradation and implications (AS)
  - Identify methods to inspect & evaluate evidence of degradations (I&E)
  - Opportunities and methods to Mitigate degradation (MT)
  - Repair or Replacement methods as available (RR)
  - Listing of Regulatory driven issues (RG)
- Approach :
  - Collection and Updating of Operating Experience
    - MRP Meetings – Operating Experience updates from plants, 3 times per year
    - EPRI Emerging Issues – as arising
  - Documentation
    - Materials Degradation Matrix (MDM) and Issues Management Tables (IMT) maintained as living documents with regular updates



EPRI Systematic Approach to Materials and Components' Aging Management

Materials Degradation Matrix
3002013781

Every component, every material, every potential degradation mechanism

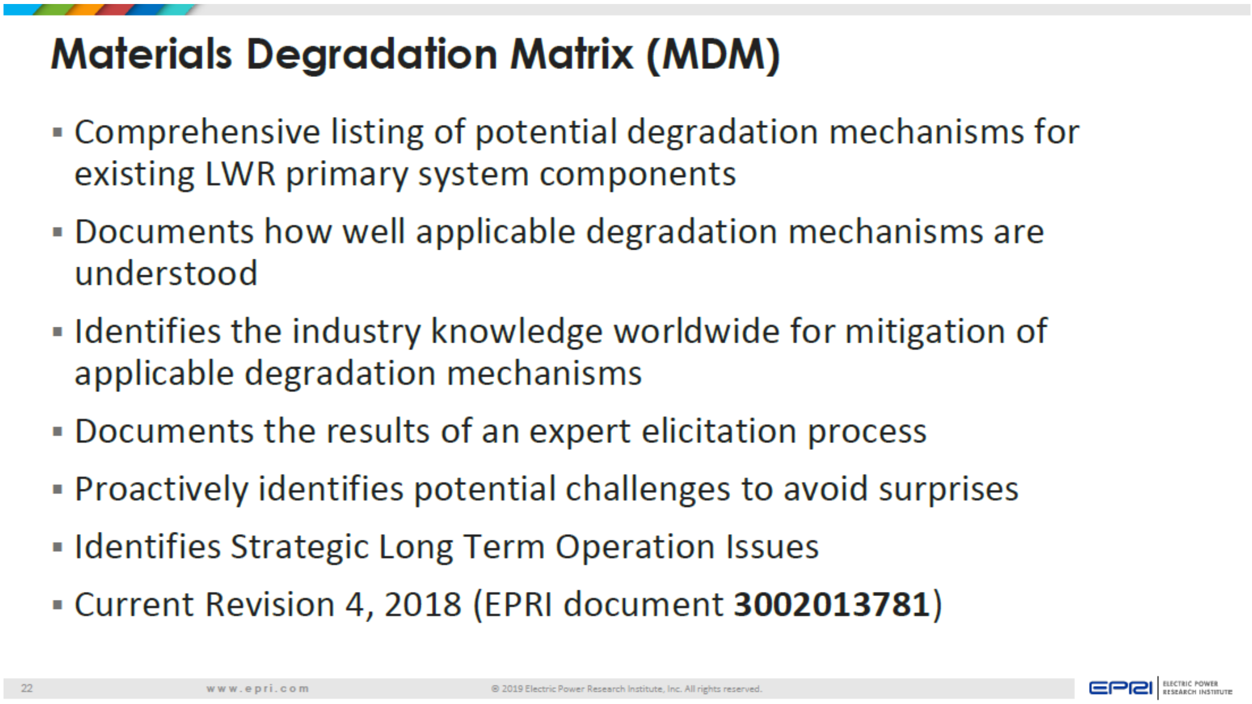
- Mapped to 80 years of operation
- Covering BWR, PWR, CANDU and VVER

Issue Management Tables
PWR: 3002000634
BWR: 3002000690
CANDU: In Progress

Every issue to resolve identified and prioritized

- Covers BWRs and PWRs
- CANDU in progress

A key objective is to help coordinate the global industry



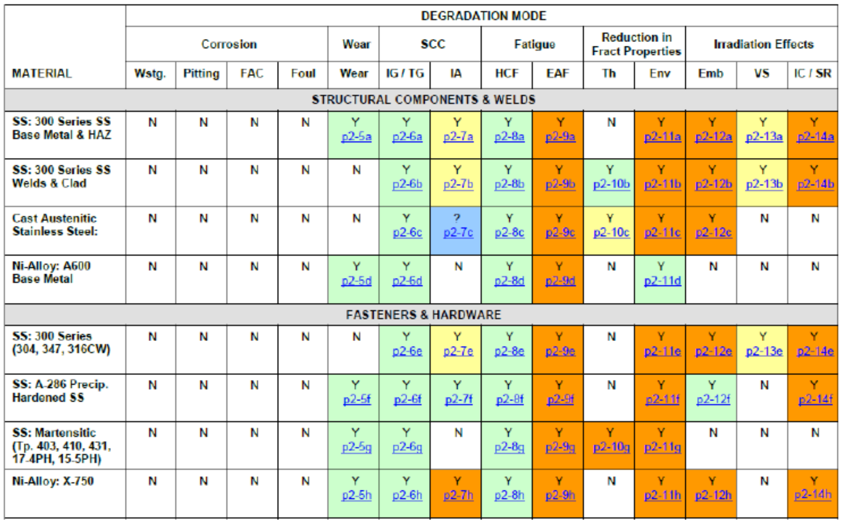
Materials Degradation Matrix (MDM)

- Comprehensive listing of potential degradation mechanisms for existing LWR primary system components
- Documents how well applicable degradation mechanisms are understood
- Identifies the industry knowledge worldwide for mitigation of applicable degradation mechanisms
- Documents the results of an expert elicitation process
- Proactively identifies potential challenges to avoid surprises
- Identifies Strategic Long Term Operation Issues
- Current Revision 4, 2018 (EPRI document **3002013781**)

MDM Results---- PWR Reactor Internals

- For groups of components…
- Matrices of materials used vs potential (standard set of) degradation modes
- Expert panel elicitation to define applicability (Y, N, ?… N/A)
- R&D status –
  - Green = sufficiently well characterized, reasonable understanding
  - Yellow = near term gaps in characterization but relevant research is in progress
  - Orange = near term gaps in characterization but relevant research is in progress
  - Blue = for all ?… = lack of data
- Notes – clarify the applicability of the mechanism and research determination

Table 4-2
PWR Reactor Vessel Internals



MATERIAL	DEGRADATION MODE														
	Corrosion				Wear	SCC		Fatigue		Reduction in Fract Properties		Irradiation Effects			
	Wstg.	Pitting	FAC	Foul	Wear	IG / TG	IA	HCF	EAF	Th	Env	Emb	VS	IC / SR
STRUCTURAL COMPONENTS & WELDS															
SS: 300 Series SS Base Metal & HAZ	N	N	N	N	Y p2-5a	Y p2-6a	Y p2-7a	Y p2-8a	Y p2-9a	N	Y p2-11a	Y p2-12a	Y p2-13a	Y p2-14a
SS: 300 Series SS Welds & Clad	N	N	N	N	N	Y p2-6b	Y p2-7b	Y p2-8b	Y p2-9b	Y p2-10b	Y p2-11b	Y p2-12b	Y p2-13b	Y p2-14b
Cast Austenitic Stainless Steel:	N	N	N	N	N	Y p2-6c	? p2-7c	Y p2-8c	Y p2-9c	Y p2-10c	Y p2-11c	Y p2-12c	N	N
Ni-Alloy: A600 Base Metal	N	N	N	N	Y p2-5d	Y p2-6d	N	Y p2-8d	Y p2-9d	N	Y p2-11d	N	N	N
FASTENERS & HARDWARE															
SS: 300 Series (304, 347, 316CW)	N	N	N	N	N	Y p2-6e	Y p2-7e	Y p2-8e	Y p2-9e	N	Y p2-11e	Y p2-12e	Y p2-13e	Y p2-14e
SS: A-286 Precip. Hardened SS	N	N	N	N	Y p2-5f	Y p2-6f	Y p2-7f	Y p2-8f	Y p2-9f	N	Y p2-11f	Y p2-12f	N	Y p2-14f
SS: Martensitic (Tp. 403, 410, 431, 17-4PH, 15-5PH)	N	N	N	N	Y p2-5g	Y p2-6g	N	Y p2-8g	Y p2-9g	Y p2-10g	Y p2-11g	N	N	N
Ni-Alloy: X-750	N	N	N	N	Y p2-5h	Y p2-6h	Y p2-7h	Y p2-8h	Y p2-9h	N	Y p2-11h	Y p2-12h	N	Y p2-14h

MDM Notes

Table 4-7 (continued)
PWR Systems Explanatory Notes

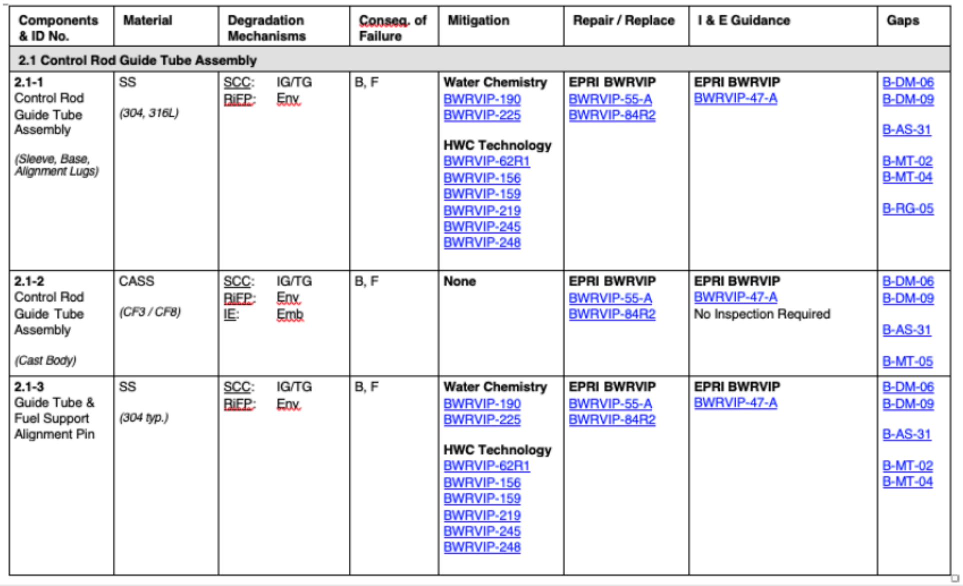
NOTE ID	EXPLANATORY NOTE
p2-6h	There has been substantial field experience with SCC of prior generation Alloy X-750 components due to sub-optimal heat treatment and designs that included high stress concentrations and poor surface finish. However, extensive laboratory testing indicates that new Alloy X-750 parts will perform satisfactorily if they meet current specification requirements with regard to chemical composition, optimized heat treatment (HTH condition), fabrication sequence, stress / strain limits, and surface finish. Older Alloy X-750 designs are being managed through inspection and replacement, sometimes with the optimized Alloy X-750 parts and sometimes with strain-hardened Type 316 stainless steel. During a routine ISI of the reactor vessel, one plant reported that seven of forty-eight Alloy X-750 clevis insert bolts showed evidence of separation between the head and the shank. Clevis insert bolt head / shank failures were evidenced by wearing of the lock bars which retain the torque of the clevis insert bolts. At one location, the clevis insert dowel pin tack welds were fractured and the dowel pin was slightly rotated and displaced deeper into the clevis insert. While not confirmed at this time, the known susceptibility of non-optimized Alloy X-750 to stress corrosion cracking and low temperature crack propagation (LTCP) is expected to be a contributor to the observed degradation. Westinghouse performed engineering evaluations of the as-found conditions and was able to justify that the system would function as required for two cycles of operation without immediate repair. At present, no additional research is considered to be needed. Screening criteria for SCC based on stress are contained in MRP-175. [R&D Status = GREEN] **References:** EPRI: MRP-88, MRP-175R1, 3002012420 (Materials Handbook) Other: [4-31]

- Defined by and pertains to the applicable matrix table cell
- Assessment of the status of the issue
- Summary of the relevant research in progress
- Hyperlinked from applicable matrix table

Issue Management Tables

- Assesses the consequences of failure
- Identifies gaps in inspection, mitigation, repair, and replacement guidance
- Prioritizes IMT gaps to direct R&D project plans of the Issue Programs
- Requires utility members engagement
- Updated 2013
  - BWRVIP-167, Rev. 3; EPRI Report 30020000690
  - MRP-205, Rev. 3; EPRI Report 3002000634

Component-Based Issue Management Tables



Components & ID No.	Material	Degradation Mechanisms	Conseq. of Failure	Mitigation	Repair / Replace	I & E Guidance	Gaps
2.1 Control Rod Guide Tube Assembly							
2.1-1 Control Rod Guide Tube Assembly *(Sleeve, Base, Alignment Lugs)*	SS *(304, 316L)*	SCC: IG/TG RIEP: Env.	B, F	**Water Chemistry** BWRVIP-190 BWRVIP-225 **HWC Technology** BWRVIP-62R1 BWRVIP-156 BWRVIP-159 BWRVIP-219 BWRVIP-245 BWRVIP-248	**EPRI BWRVIP** BWRVIP-55-A BWRVIP-84R2	**EPRI BWRVIP** BWRVIP-47-A	B-DM-06 B-DM-09 B-AS-31 B-MT-02 B-MT-04 B-RG-05
2.1-2 Control Rod Guide Tube Assembly *(Cast Body)*	CASS *(CF3 / CF8)*	SCC: IG/TG RIEP: Env. IE: Emb	B, F	**None**	**EPRI BWRVIP** BWRVIP-55-A BWRVIP-84R2	**EPRI BWRVIP** BWRVIP-47-A No Inspection Required	B-DM-06 B-DM-09 B-AS-31 B-MT-05
2.1-3 Guide Tube & Fuel Support Alignment Pin	SS *(304 typ.)*	SCC: IG/TG RIEP: Env.	B, F	**Water Chemistry** BWRVIP-190 BWRVIP-225 **HWC Technology** BWRVIP-62R1 BWRVIP-156 BWRVIP-159 BWRVIP-219 BWRVIP-245 BWRVIP-248	**EPRI BWRVIP** BWRVIP-55-A BWRVIP-84R2	**EPRI BWRVIP** BWRVIP-47-A	B-DM-06 B-DM-09 B-AS-31 B-MT-02 B-MT-04

Background – R&D Gap Categories

- Degradation Mechanism Understanding (DM) – Fundamental materials knowledge gaps (*understanding the degradation mechanisms affecting primary systems components*)
- Assessment (AS) – Gaps related to understanding the detrimental impacts of degradation on component performance and managing these effects to support continued safe and reliable operation
- Mitigation (MT) – Technology gaps related to preventing or mitigating the occurrence or progression of degradation mechanisms (*encompasses not only water chemistry based mitigation, but other methods as well – e.g., surface stress improvement*)
- Inspection & Evaluation (I&E) – Technology gaps related to inspection / NDE capabilities and uncertainties
- Repair / Replacement (RR) – Technology gaps related to repair / replacement capabilities
- Regulatory (RG) – Issues driven by regulatory requirements or by regulator imposed requirements

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Issue Management Gap Tables

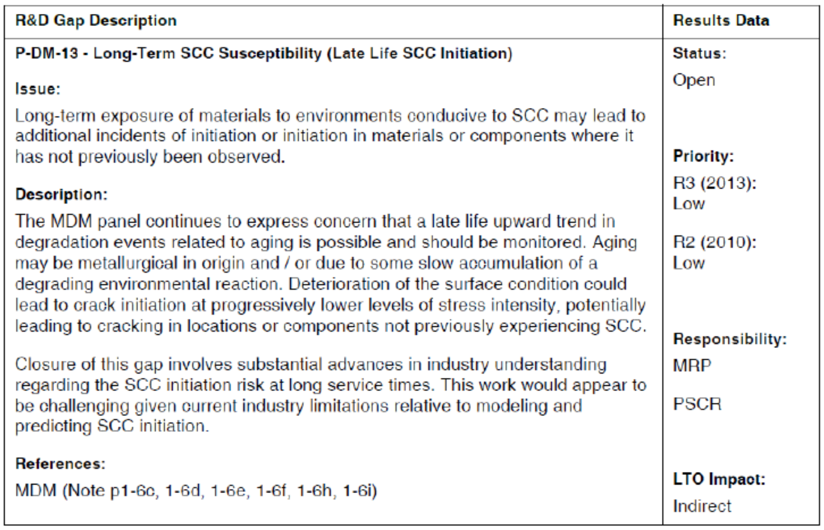
- Builds on Materials Degradation Matrix
- Combine with evaluations of Consequences of Failure of Components (e.g. MRP-156 & MRP-157)
- Identify gaps in
  - Understanding of the degradation method
  - Assessment methods and capabilities
  - Mitigation methods
  - Inspection and evaluation methods and capabilities
  - Repair and replacement availability
  - Ability to respond to regulatory issues
- Prioritize the gaps
  - ***High, Medium, Low***
- Work projects based on issues' priorities
- Update the tables in cycles
  - Close gaps as knowledge and capabilities are developed
  - New gaps as OE, licensing, regulatory questions arise

Table 3-1 R&D Gap Priority Results

Gap ID No.	Gap Description
Degradation Mechanism Understanding Gaps (Table 3-2)	
P-DM-09	Environmental Effects on Fracture Resistance
P-DM-10	Thermal Embrittlement of Low-Alloy Pressure Vessel Steels
P-DM-11	SCC (and Thermal Aging) of CASS Pressure Boundary Components
P-DM-12	Increased Fastener SCC Susceptibility due to Long-Term Aging
P-DM-13	Long-Term SCC Susceptibility (Late Life SCC Initiation)
P-DM-14	Long-Term Stability of Surface Stress Improvement Mitigations
P-DM-15	Thermal Embrittlement of Martensitic Stainless Steels
P-DM-16	Thermal Embrittlement of Martensitic Stainless Steels (SG Tube Support Plates)

Issue Management Gap Tables – Open Gaps

Table 3-2 Degradation Mechanism Understanding Gaps (Continued)

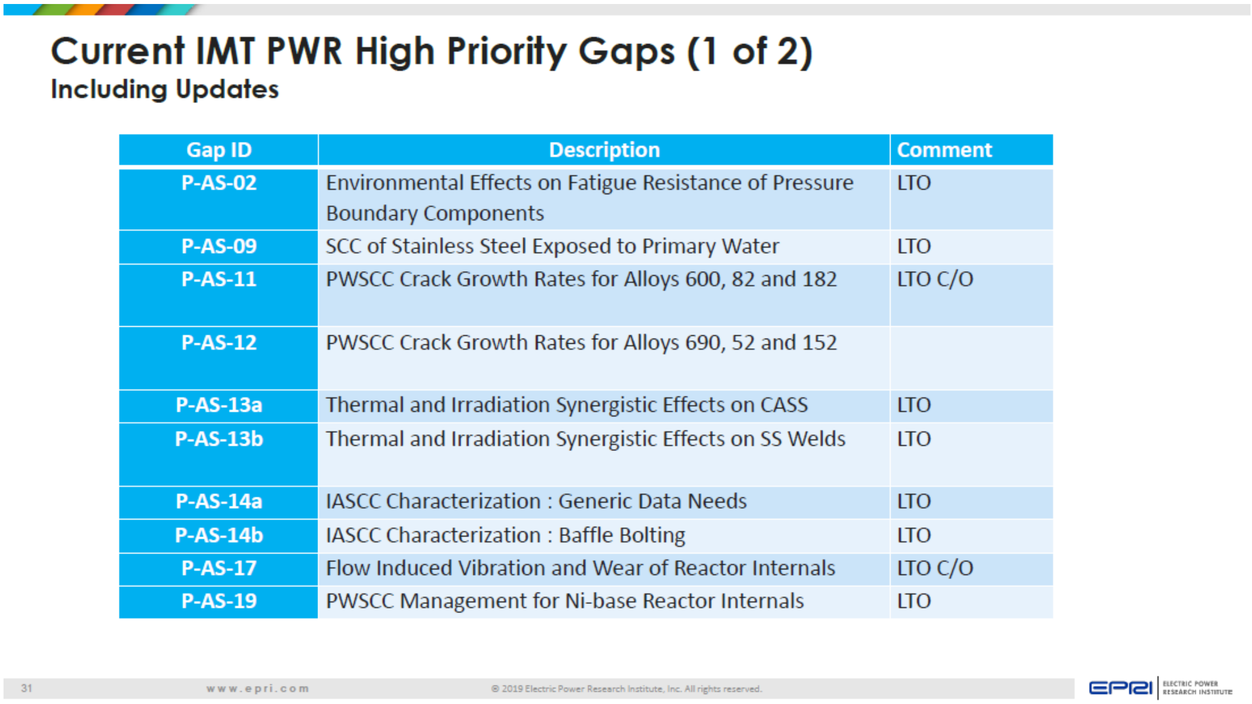


R&D Gap Description	Results Data
P-DM-13 - Long-Term SCC Susceptibility (Late Life SCC Initiation) **Issue:** Long-term exposure of materials to environments conducive to SCC may lead to additional incidents of initiation or initiation in materials or components where it has not previously been observed. **Description:** The MDM panel continues to express concern that a late life upward trend in degradation events related to aging is possible and should be monitored. Aging may be metallurgical in origin and / or due to some slow accumulation of a degrading environmental reaction. Deterioration of the surface condition could lead to crack initiation at progressively lower levels of stress intensity, potentially leading to cracking in locations or components not previously experiencing SCC. Closure of this gap involves substantial advances in industry understanding regarding the SCC initiation risk at long service times. This work would appear to be challenging given current industry limitations relative to modeling and predicting SCC initiation. **References:** MDM (Note p1-6c, 1-6d, 1-6e, 1-6f, 1-6h, 1-6i)	**Status:** Open **Priority:** R3 (2013): Low R2 (2010): Low **Responsibility:** MRP PSCR **LTO Impact:** Indirect

- Clear statement of the gap and consequence
- Description of the issues related to the gap
- References to MDM notes
- Priority – ***Low, Medium, or High***
- History of revisions previous priority, newly opened
- (Closed items also identified in EPRI documents for completeness)

PWR Issue Management Gap Tables

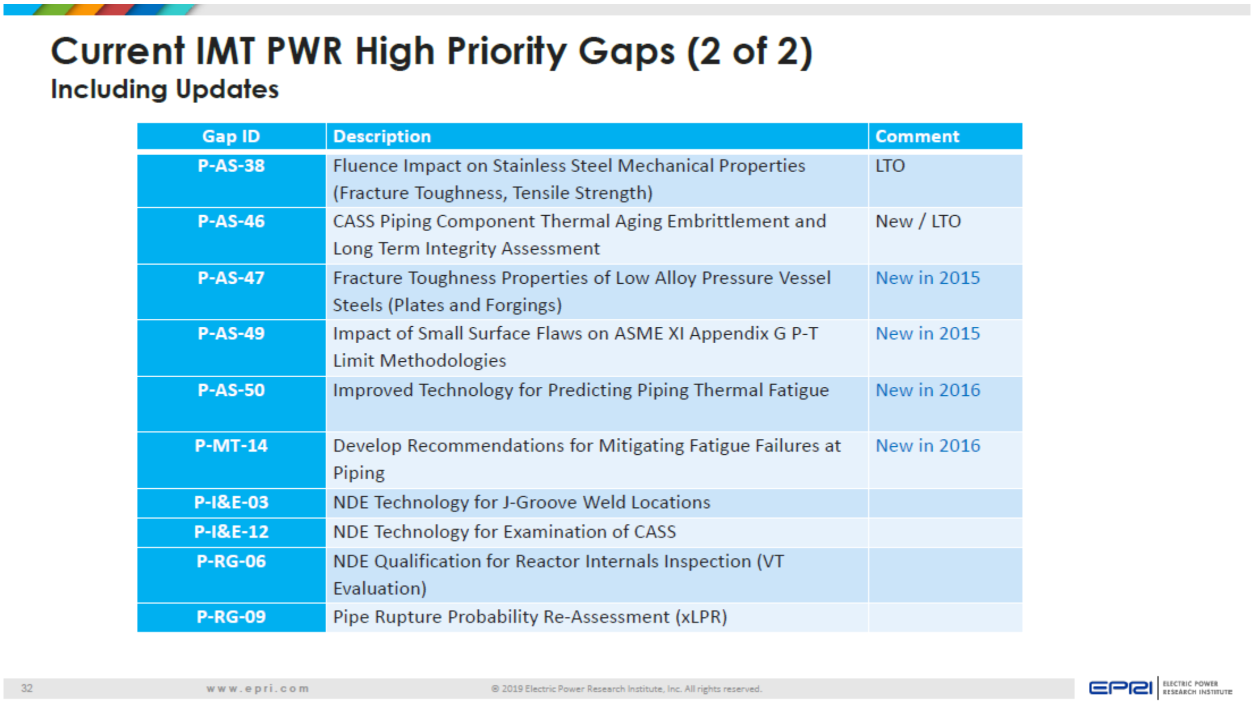
- Identifies and prioritizes "gaps" by reviewing impact at a subcomponent level
- Focuses efforts where most needed
- Used to set priorities and funding (High, Medium & Low) for research projects
- Used to inform stakeholders of key areas that require efforts
  - EPRI Program Members
  - Collaborators – MAI, CRIEPI, US-DoE & National Labs
  - Regulatory Agencies – US-NRC, ENSI, IAEA



Current IMT PWR High Priority Gaps (1 of 2)

Including Updates

Gap ID	Description	Comment
P-AS-02	Environmental Effects on Fatigue Resistance of Pressure Boundary Components	LTO
P-AS-09	SCC of Stainless Steel Exposed to Primary Water	LTO
P-AS-11	PWSCC Crack Growth Rates for Alloys 600, 82 and 182	LTO C/O
P-AS-12	PWSCC Crack Growth Rates for Alloys 690, 52 and 152	
P-AS-13a	Thermal and Irradiation Synergistic Effects on CASS	LTO
P-AS-13b	Thermal and Irradiation Synergistic Effects on SS Welds	LTO
P-AS-14a	IASCC Characterization : Generic Data Needs	LTO
P-AS-14b	IASCC Characterization : Baffle Bolting	LTO
P-AS-17	Flow Induced Vibration and Wear of Reactor Internals	LTO C/O
P-AS-19	PWSCC Management for Ni-base Reactor Internals	LTO



Current IMT PWR High Priority Gaps (2 of 2)

Including Updates

Gap ID	Description	Comment
P-AS-38	Fluence Impact on Stainless Steel Mechanical Properties (Fracture Toughness, Tensile Strength)	LTO
P-AS-46	CASS Piping Component Thermal Aging Embrittlement and Long Term Integrity Assessment	New / LTO
P-AS-47	Fracture Toughness Properties of Low Alloy Pressure Vessel Steels (Plates and Forgings)	New in 2015
P-AS-49	Impact of Small Surface Flaws on ASME XI Appendix G P-T Limit Methodologies	New in 2015
P-AS-50	Improved Technology for Predicting Piping Thermal Fatigue	New in 2016
P-MT-14	Develop Recommendations for Mitigating Fatigue Failures at Piping	New in 2016
P-I&E-03	NDE Technology for J-Groove Weld Locations	
P-I&E-12	NDE Technology for Examination of CASS	
P-RG-06	NDE Qualification for Reactor Internals Inspection (VT Evaluation)	
P-RG-09	Pipe Rupture Probability Re-Assessment (xLPR)	

Ongoing EPRI Research Programs Address Issue Gaps – Just some of the current projects

- SCC of stainless steels in off-chemistry environments
- Mechanistic understanding of SCC in stainless steels
- IASCC of irradiated stainless steels in elevated LiOH and KOH balanced PWR coolant
- Re-examination of irradiation effects and void swelling in irradiated stainless steels
- Fluence impact on fracture toughness of irradiated stainless steels
- Crack growth testing of irradiated stainless steels
- Updating of reactor internals guidelines (MRP-227 process) documentation
- Tracking and correlation of RV internals I&E guidelines
- Assessment of long time thermal effects on embrittlement of low alloy steels
- Assessment of C segregation effects on fracture resistance of low alloy steels
- PWR Supplementary Surveillance Program to re-irradiated pre-exposed surveillance materials
- Improved reactor vessel degradation modeling – use of master curve approach
- Assessment of environmental effects on fatigue
- PWSCC crack initiation and crack growth data correlations in Alloy 600 and Alloy 690
- Potential LRO and its effects on PWSCC of Alloy 690
- Effect of cold work and welding on PWSCC of Alloy 690
- Risk based pipe rupture methodology, xLPR

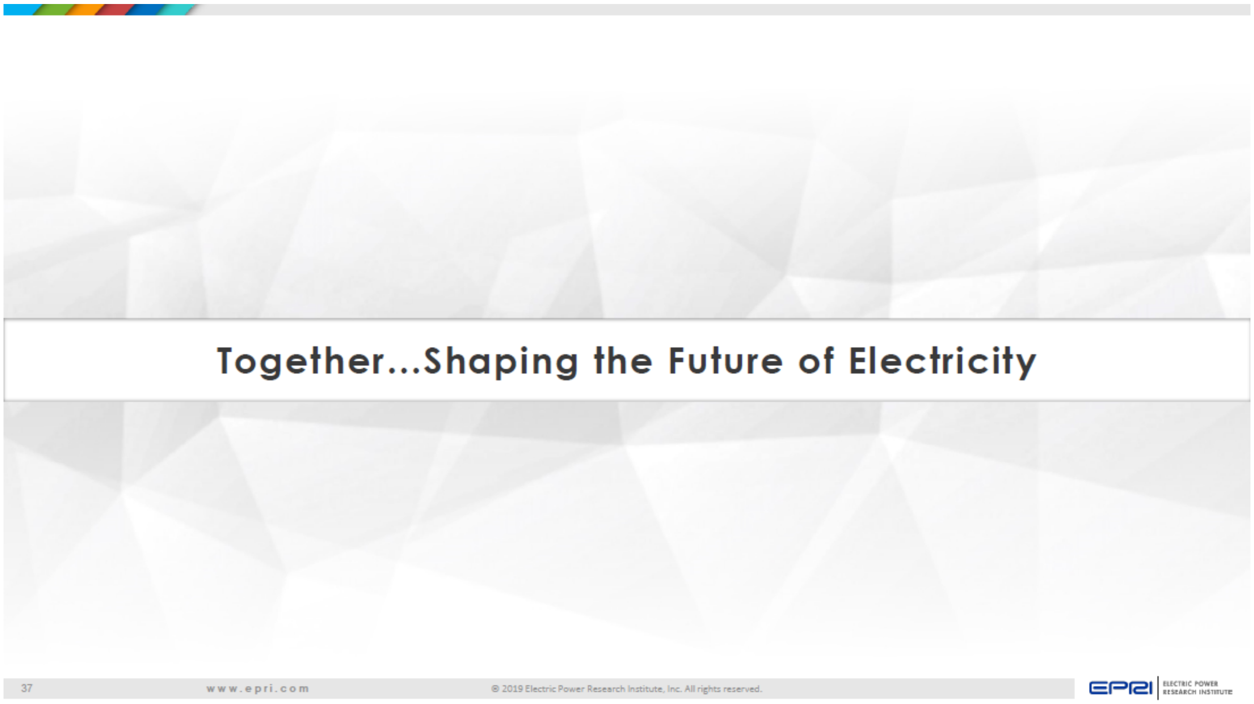
Initiative Accomplishments

- Integrated industry strategic plan for materials
- Achieved a high level of industry integration, coordination, alignment, and communication on material issues
- Established a process for prioritizing projects, budgets, and planning
- Predictable funding for materials R&D – clear definition of ***NEEDED*** projects
- Engaged INPO as an active participant
- Defined expectations and protocols for industry actions upon discovery of an emergent issue
- Established consistent process for deviations and communication with NRC
- Executive level interactions between industry and senior NRC management
- Successful at closing materials issues and gaps
- Fewer unexpected materials related transients

Summary

- NEI 03-08 set an expectation for proactive materials aging and degradation management
- The strategic approach utilizes the MDM and IMTs
- Based on significant and continually updated plant OE inputs
- EPRI's MDM and IMTs are living documents that reflect:
  - Operating experience
  - Updates from research programs
  - Status and priority of issues
- The MDM and IMTs are effective tools for the MRP and other programs to use in development of strategic plans to address needed knowledge generation.
- IMT prioritized gaps drive the EPRI funded research programs.

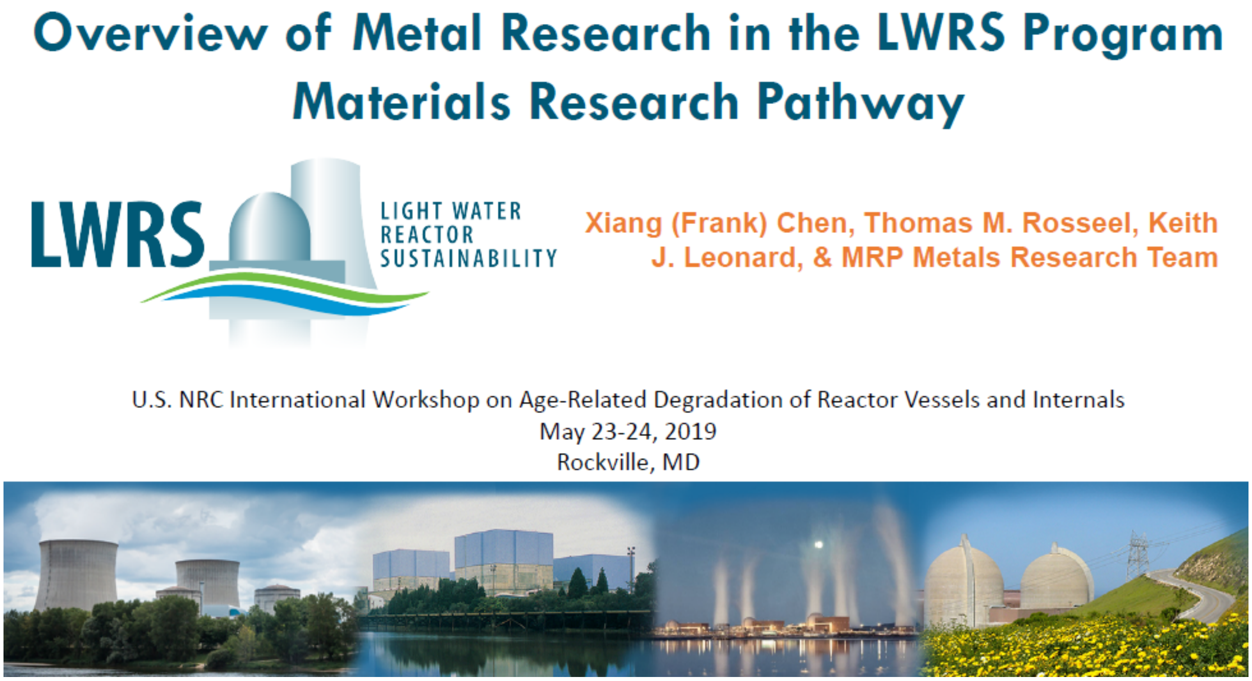
Questions



Together...Shaping the Future of Electricity

37 www.epri.com

B.4 Overview of Metals Research in LWRS Program MRP (F. Chen)



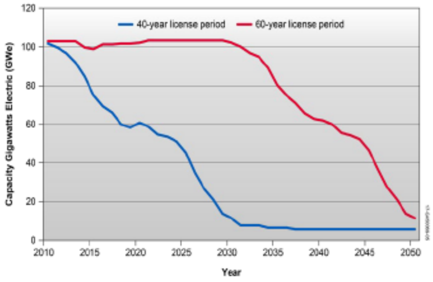
Overview of Metal Research in the LWRS Program Materials Research Pathway

Xiang (Frank) Chen, Thomas M. Rosseel, Keith J. Leonard, & MRP Metals Research Team

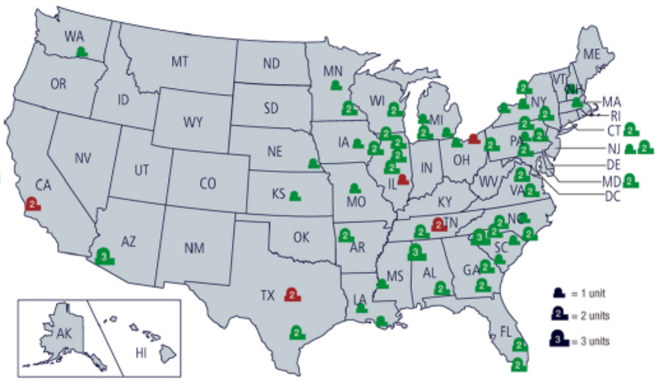
U.S. NRC International Workshop on Age-Related Degradation of Reactor Vessels and Internals
May 23-24, 2019
Rockville, MD

Extended operations of the existing commercial power generation reactor fleet is in the U.S. national interest

- Our LWRs are a national asset
  - ~100 GWe of low-carbon generation
  - Low-cost, reliable generation
  - Energy diversity
  - Nearly $1T replacement cost
- Even with first 20-year extension, nearly all plants will reach the end of their 60-year license between 2030 and 2055



License Renewals Granted for Operating Nuclear Power Reactors



Licensed to Operate (98)

▲ Original License (8) ▲ License Renewal Granted (90)

Note: The NRC has issued a total of 94 license renewals; four of these units have permanently shut down. Data are as of March 2019.

U.S.NRC
Data as of March 2019

LWRS Program Goals and Objectives

The mission of the Light Water Reactor Sustainability Program (LWRS) is to develop the scientific basis, and science-based methodologies and tools, for the safe economical long-term operation of the nation's high-performing fleet of commercial nuclear energy facilities

- **Objectives**
  - Provide science and technology-based solutions to industry to overcome the current labor-intensive business model and associated practices
  - Manage the aging of systems, structures, and components so nuclear power plants can continue to operate safely and cost effectively
- **Pathways**
  - Materials Research
  - Plant Moderation
  - Risk-informed Systems Analysis

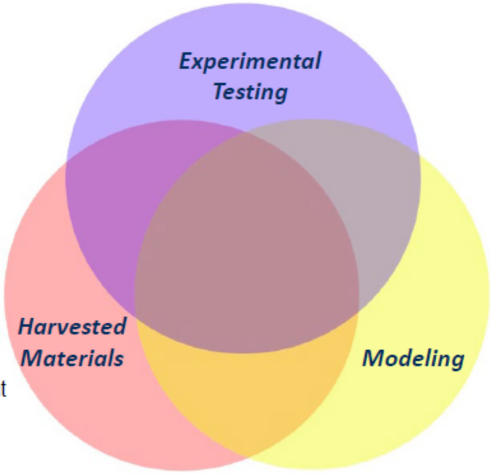
Nine Mile Point (courtesy Exelon)

The LWRS program is the primary U.S. DOE program for light water reactor research, development and demonstration

LWRS LIGHT WATER REACTOR SUSTAINABILITY

Materials Research Pathway (MRP)

- **Objectives**
  - Develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants
- **Approach: guided by the "5M"**
  - Measurements of degradation
  - Mechanisms of degradation
  - Modeling and simulation
  - Monitoring
  - Mitigation strategies
- **Research benefits**
  - Understanding which components are susceptible to certain forms of degradation, and their predictive behavior, will permit more focused component inspections, component replacements, and more detailed regulatory guideline
  - The R&D products will be used by utilities, industry groups, and regulators
- **Partners**
  - EPRI, Westinghouse, PWROG, CRIEPI, Rolls Royce, Exelon, U.S. NRC



LWRS LIGHT WATER REACTOR SUSTAINABILITY

Materials Research Pathway Metal-Related Project Portfolio

Reactor Pressure Vessel (RPV)

Mitigation Technologies

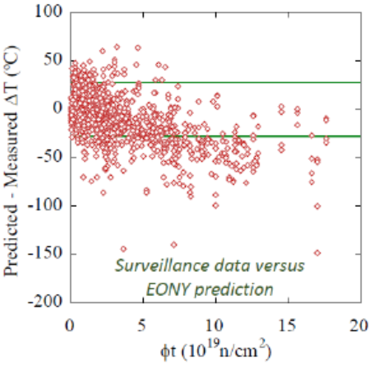
Harvesting and Integrated Research

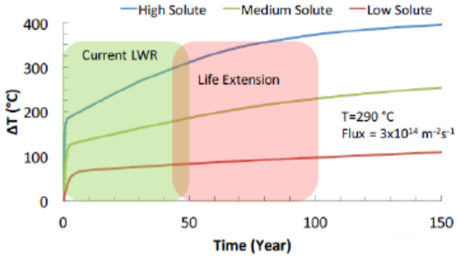
Core Internals and Piping

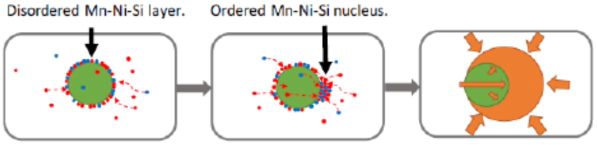
Other Activities

LWRS LIGHT WATER REACTOR SUSTAINABILITY Reactor Pressure Vessel (RPV)

- **ATR-2: UCSB, Odette**
  - Expand database (microstructure / mechanical properties) on effects of fluence, flux, temperature over a large range of alloy compositions
- **Modeling of Cu and MnNiSi precipitates: U. Wisconsin, Morgan**
  - Predict the Cu and MnNiSi formation for a wide range of compositions, neutron fluxes, and temperatures
  - Provide information on the formation mechanism of Cu-core-MnNiSi-appendage microstructure







Disordered Mn-Ni-Si layer.

Ordered Mn-Ni-Si nucleus.

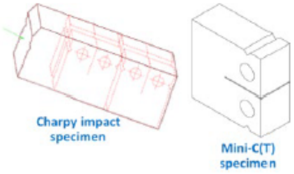
Cu-rich precipitate with disordered Mn-Ni-Si coating layer.

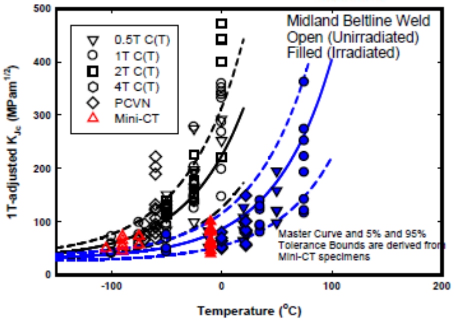
Formation of an ordered Mn-Ni-Si nucleus.

Later stage: solute atoms flow to the ordered Mn-Ni-Si phase both through matrix and Cu cluster.

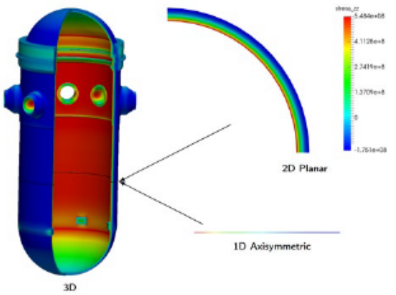
LWRS LIGHT WATER REACTOR SUSTAINABILITY Reactor Pressure Vessel (RPV)

- **Mini-compact tension specimen: ORNL, Sokolov**
  - Develop the testing technology for determining Master Curve fracture toughness for RPV steels





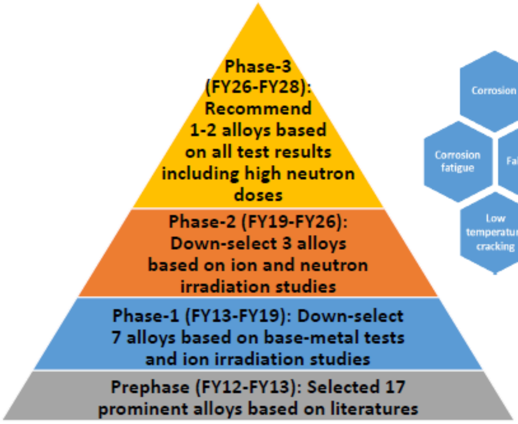
- **Grizzly multi-physics simulation: INL, Spencer**
  - Develop a simulation tool to predict the progression of aging mechanisms and their effects on integrity of multiple critical nuclear power plant components (RPVs, concrete structures, etc.)



Results of 1D axisymmetric, 2D planar, and 3D Grizzly models of the global response of an RPV at a point in time during a PTS event

Mitigation Technologies

- **Advanced replacement alloys: ORNL/Tan, U. Michigan/Was, EPRI ARRM program, KAPL, GE, PNNL, INL**
  - Down-select and develop advanced radiation resistant materials (ARRM) for core internal support components and fasteners as replacement materials in current LWRs and new materials in future reactors



Phase-3 (FY26-FY28): Recommend 1-2 alloys based on all test results including high neutron doses

Phase-2 (FY19-FY26): Down-select 3 alloys based on ion and neutron irradiation studies

Phase-1 (FY13-FY19): Down-select 7 alloys based on base-metal tests and ion irradiation studies

Prephase (FY12-FY13): Selected 17 prominent alloys based on literatures

Non-rad & Rad tests

Candidate Alloys

Low Strength

316L (standard)

- 310
- ~~800~~ *IASCC*
- Grade 92 (NF616)
- ~~HT9~~ *Fracture toughness*
- ~~C22~~ *Radiation-induced hardening*
- ~~625~~ *IASCC*
- 690
- ~~Zr-2.5Nb~~ *Accident tolerance concerns*
- ~~Ti alloys~~ *Fracture toughness*

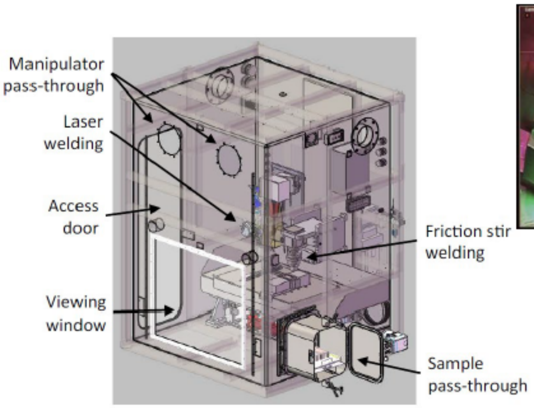
High Strength

X-750 (standard)

- ~~439~~ *Fracture toughness / processing*
- ~~625 plus~~ *IASCC*
- ~~625 DA~~ *IASCC and stability*
- 725
- 718 (heat A)
- ~~14WYT~~ *Fracture toughness / fabr.*
- ~~Hi-Cr, Al ODS~~ *Examined under FCRD*

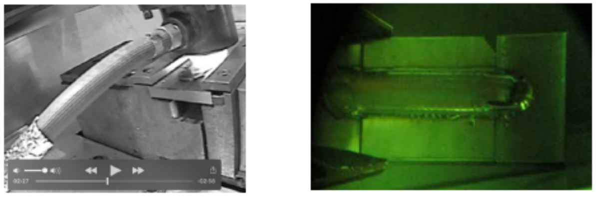
Mitigation Technologies

- **Welding repair: ORNL (Feng et al.) & EPRI (Frederick et al.)**
  - Develop advanced welding technologies to weld highly irradiated materials while avoiding helium induced cracking



Irradiated materials welding facility at ORNL

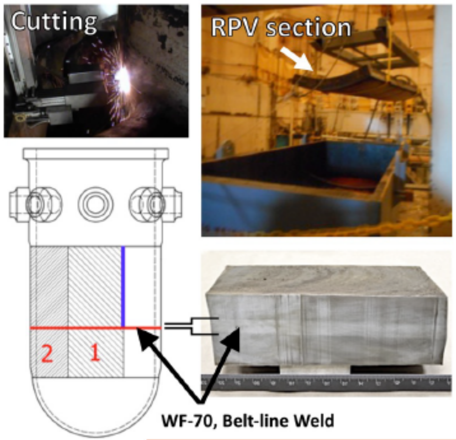
Laser welding of irradiated 304L

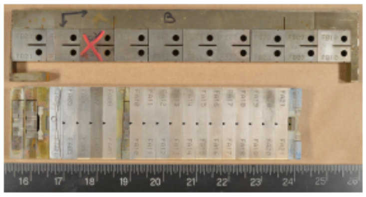


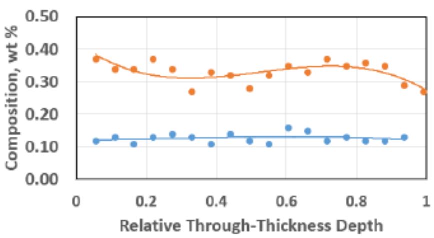
Friction stir welding of irradiated 304L

Harvesting and Integrated Research

- **Zion RPV: ORNL, Rosseel & Sokolov**
  - Evaluate radiation damage models and compare results to surveillance and test reactor experiments
  - Evaluate attenuation and through wall variations in properties and composition of the base metal and the belt-line





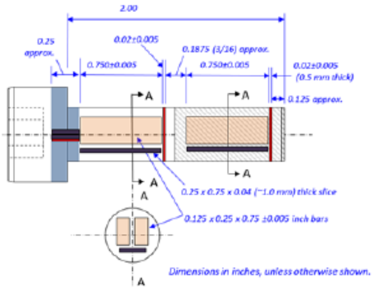


Harvesting → Machining → Testing

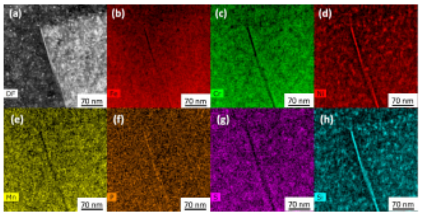
Harvesting and Integrated Research

- **Baffle former bolts : ORNL, Chen**
  - Provide critical information for evaluating end of life microstructure and properties as a benchmark of international models developed for predicting radiation-induced swelling, segregation, precipitation

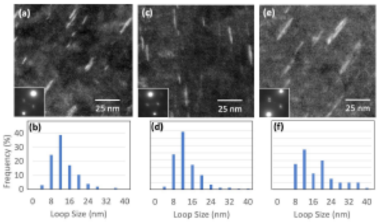
Bolt #	Fluence (10^{22} n/cm^2, E>1 MeV)/Estimated dpa		
	Head	Mid-shank	Mid-thread
4412	2.78/41	2.27/34	1.46/22
4416*	1.91/29	1.56/23	1.00/15



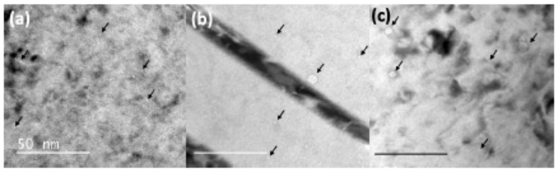
Baffle former bolt machining plan



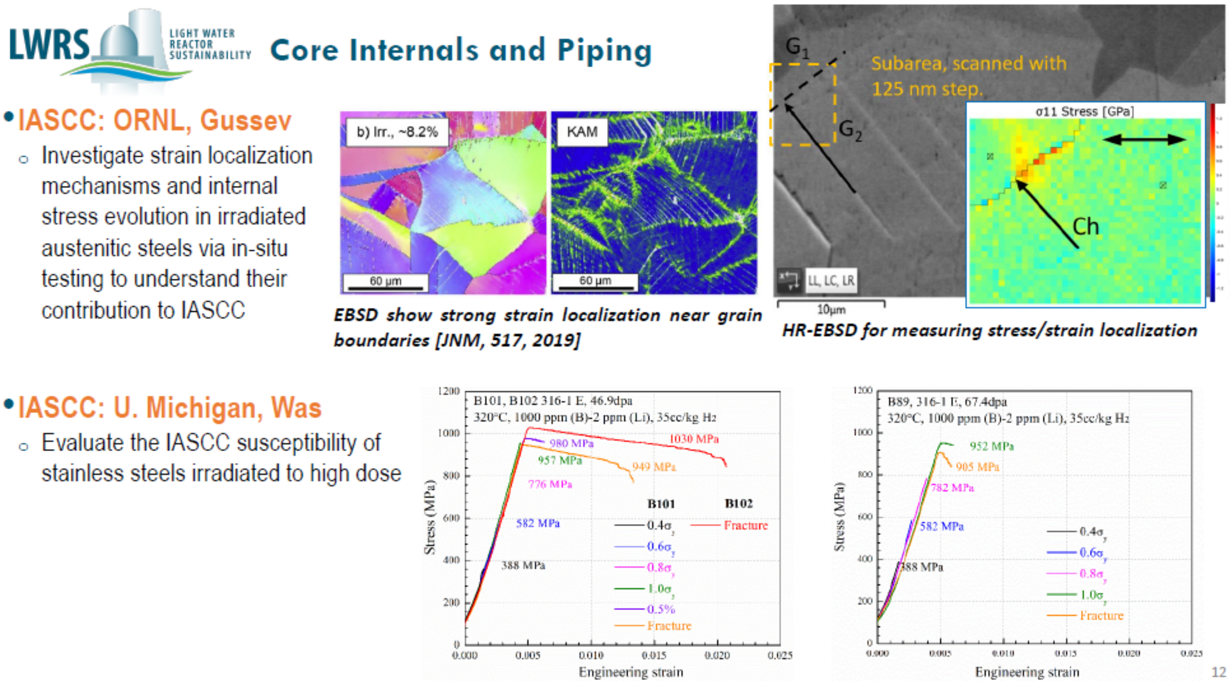
STEM/EDS mapping showing RIS



Dislocation loop characterization



Cavity characterization



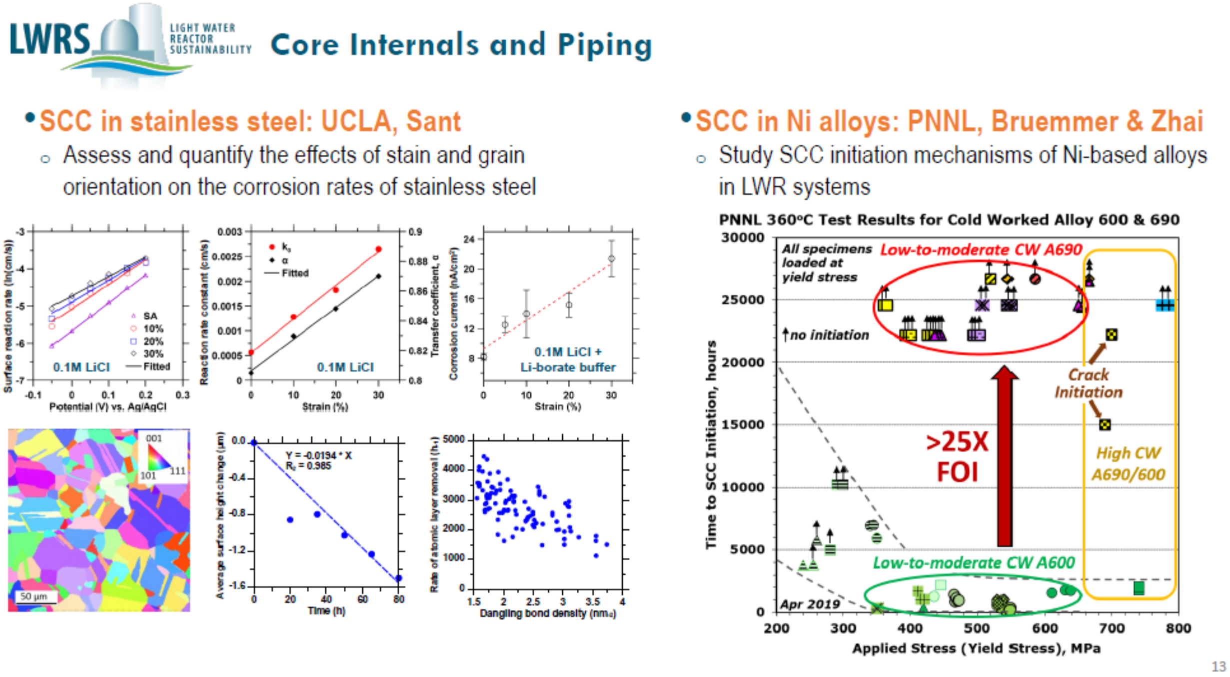
Core Internals and Piping

- **IASCC: ORNL, Gussev**
  - Investigate strain localization mechanisms and internal stress evolution in irradiated austenitic steels via in-situ testing to understand their contribution to IASCC

EBSD show strong strain localization near grain boundaries [JNM, 517, 2019]

HR-EBSD for measuring stress/strain localization

- **IASCC: U. Michigan, Was**
  - Evaluate the IASCC susceptibility of stainless steels irradiated to high dose



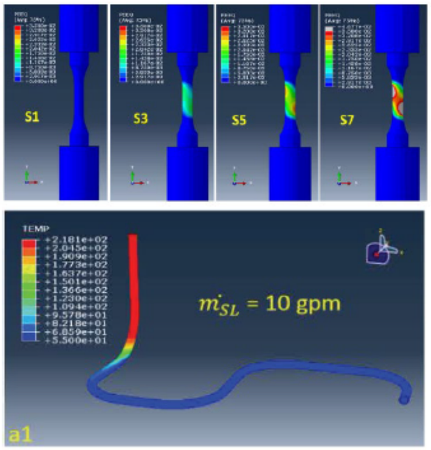
Core Internals and Piping

- **SCC in stainless steel: UCLA, Sant**
  - Assess and quantify the effects of stain and grain orientation on the corrosion rates of stainless steel

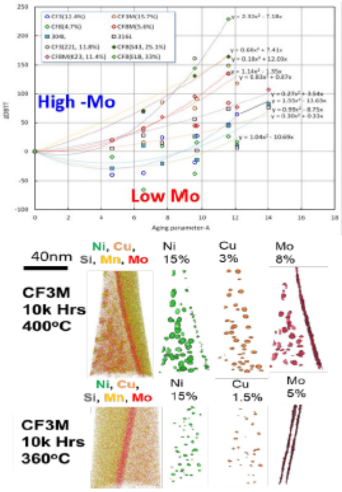
- **SCC in Ni alloys: PNNL, Bruemmer & Zhai**
  - Study SCC initiation mechanisms of Ni-based alloys in LWR systems

Core Internals and Piping

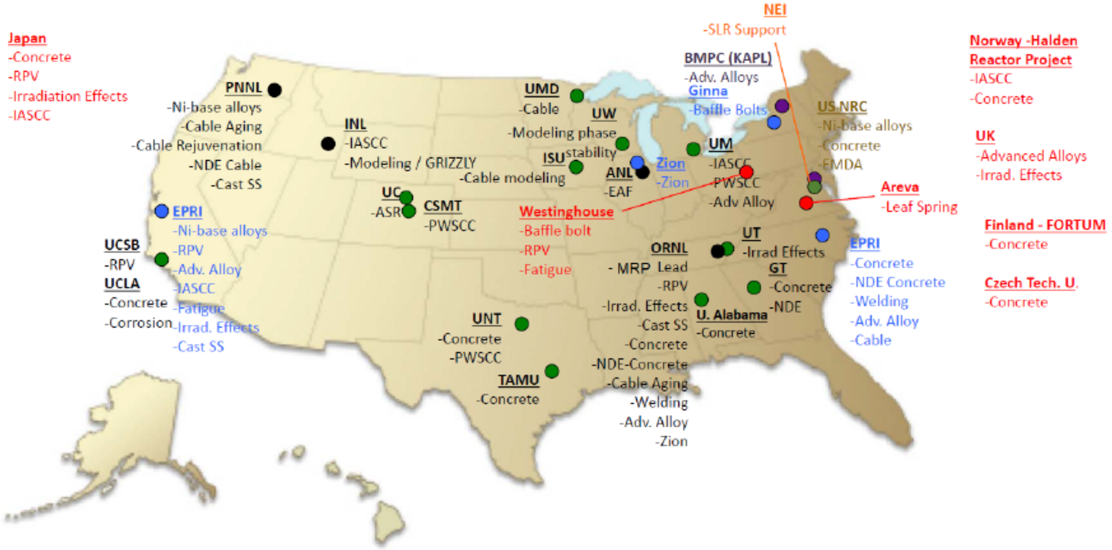
- **Environmentally assisted fatigue: ANL, Natesan & Mohanty**
  - Combine testing and modeling for LTO life estimation of primary-pressure-boundary reactor components with dissimilar weld nozzles



- **Cast stainless steel aging: PNNL, Byun**
  - Study thermal degradation of mechanical properties of casting austenitic stainless steels



Other Activities



LWRS LIGHT WATER REACTOR SUSTAINABILITY

Sustaining National Nuclear Assets

http://lwrs.inl.gov

Materials Research Pathway Contact:
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Xiang (Frank) Chen, Pathway Deputy Lead, 865-574-5058, chenx2@ornl.gov

B.5 Overview of Safety Research on Metal Aging Due to Neutron Irradiation in S/NRA/R (K. Arai)

NRA, Japan
Nuclear Regulation Authority

Overview of Safety Research on Metal Aging due to Neutron Irradiation in S/NRA/R

International Workshop on Age-Related Degradation of
Reactor Vessels and Internals

U.S. NRC Headquarters, North Bethesda, MD
May 23-24, 2019

K. Arai, K. Sakamoto, T. Hojo and M. Ikeda
Division of Research for Reactor System Safety
Regulatory Standard and Research Department
Secretariat of Nuclear Regulation Authority (S/NRA/R)

Previous research programs[(*)] on metal aging due to neutron irradiation

No.	projects	Outline
1	Prediction of irradiation embrittlement for highly irradiated reactor vessel steels (2005-2010)	- Verification of the domestic prediction formula of irradiation embrittlement in the high fluence region through conducting tests using materials with accelerated irradiation
2	Repair welding technology of irradiated materials (1997-2004)	- Verification of the integrity of irradiated stainless-steel welding of reactor internals ✓ Relationship between helium content, welding heat input and crack occurrence - Proposal of technical guides to repair the welding of reactor internals

(*)Note: These programs were conducted by Japan Nuclear Energy Safety organization (JNES).

Previous research programs[(*)] on metal aging due to neutron irradiation (Cont'd)

No.	projects	Outline
3	Evaluation of irradiation assisted stress corrosion cracking (2000-2008)	[BWR] - Evaluation of IASCC susceptibility in L-grade SSs - IASCC crack-growth database of neutron irradiated L-grade SSs - Development of crack-growth rate disposition curves for BWR normal and hydrogen water chemistry (IASCC evaluation guide) [PWR] - IASCC initiation data of baffle former bolt (BFB) in PWR primary water (<70dpa) - Development of a lifetime evaluation method for BFB (IASCC evaluation guide)
4	Evaluation of neutron irradiation effect on SCC crack growth of L-grade stainless steel (2007-2013)	[BWR] - Evaluation of the synergy effect of low neutron fluence and weld hardening on SCC crack propagation - Improvement of SCC evaluation method in terms of neutron irradiation

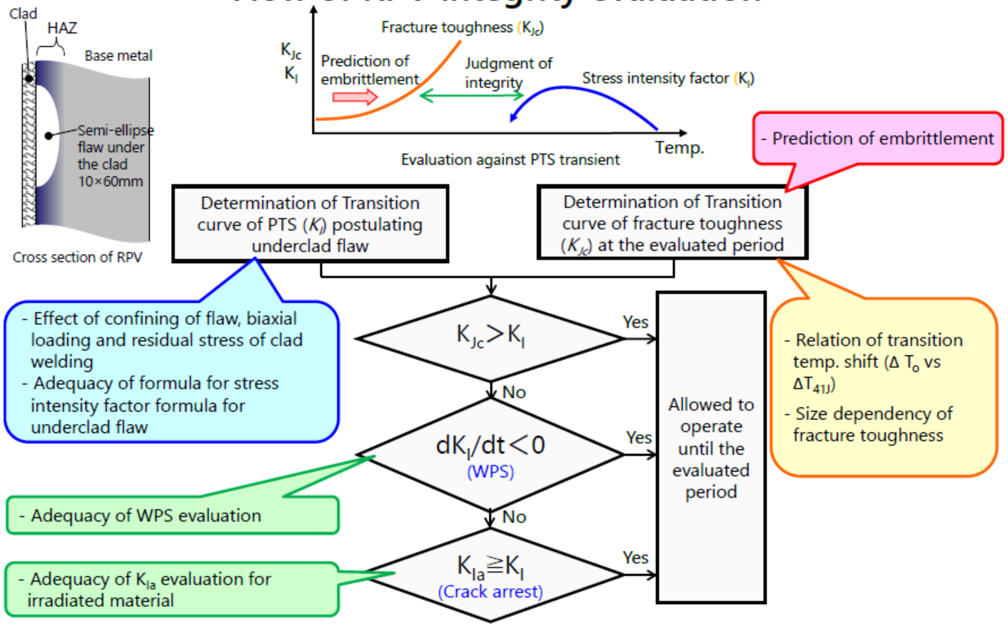
(*)Note: These programs were conducted by Japan Nuclear Energy Safety organization (JNES).

Current research programs on metal aging due to neutron irradiation

- Reactor Pressure Vessel (RPV) embrittlement
- Probabilistic Fracture Mechanics (PFM)

RPV Embrittlement

Flow of RPV integrity evaluation



- Prediction of embrittlement

Determination of Transition curve of PTS (K_I) postulating underclad flaw

Determination of Transition curve of fracture toughness (K_{Jc}) at the evaluated period

- Effect of confining of flaw, biaxial loading and residual stress of clad welding
- Adequacy of formula for stress intensity factor formula for underclad flaw

- Relation of transition temp. shift (ΔT_o vs ΔT_{41J})
- Size dependency of fracture toughness

$K_{Jc} > K_I$ — Yes / No

$dK_I/dt < 0$ (WPS) — Yes / No

$K_{Ia} \geqq K_I$ (Crack arrest) — Yes

Allowed to operate until the evaluated period

- Adequacy of WPS evaluation

- Adequacy of K_{Ia} evaluation for irradiated material

Work items

- Pressurized Thermal Shock (PTS) experiments with cladded specimens (JFY2016-2019)
- Warm Pre-Stress (WPS) effect (JFY2014-2019)
- Correlation of transition temp. shift (JFY2014-2019)
- Size dependency of fracture toughness (JFY2014-2019)
- Prediction of irradiation embrittlement (JFY2017-2019)

PTS experiments

Background

- The location of the postulated flaw was changed from an inner surface to an embedded flaw under the clad in the newly proposed code(JEAC4206-2016).
- The crack tip constraint of the postulated underclad flaw is lower than that of the test specimen, making the critical load lower.
- Biaxial load should be considered in the actual plant.
- WPS effect under biaxial loading has not been verified in a large test specimen.

Objective

- To confirm the effect of biaxial loading and underclad flaw

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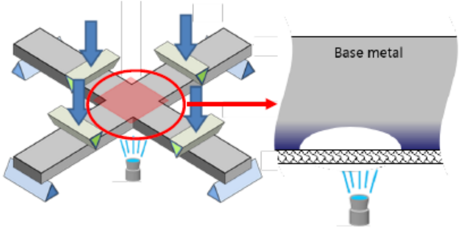
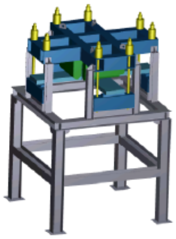
PTS experiments (Cont'd)

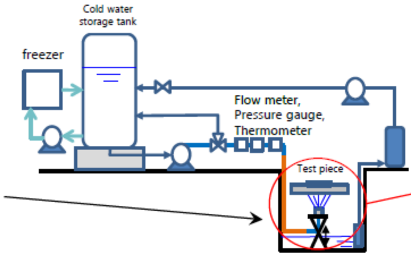
- Biaxial bending tests are conducted under simulating PTS conditions using low-toughness material.
- Test parameters such as flaw size, load, and cooling rate are determined via FEM analysis.



Large scale biaxial test specimen
- Length: 3800mm
- Thickness: Base metal 150mm, Clad 5.5mm



Load machine

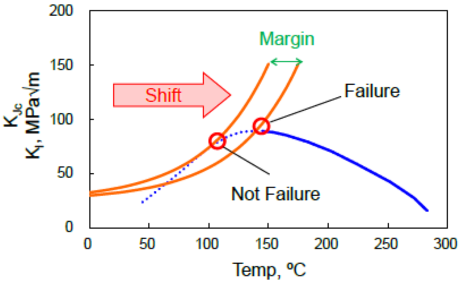


Equipment schematic

Study on WPS effect

Background

- The warm pre-stress (WPS) effect corresponds to no occurrence of crack propagation after pre-stressing in the process that the load is decreased as the temperature is decreased.
- Newly proposed code (JEAC4206-2016) introduces WPS consideration.
- The WPS effect on flaws postulated in the actual plant (underclad, post irradiated) is unknown.



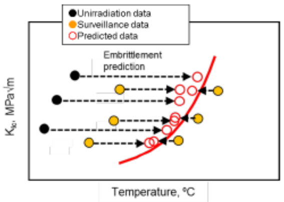
Objectives

- To confirm the adequacy of the proposed code through the experiments using irradiated material
- To develop a method of analysis to evaluate the WPS effect

Correlation of transition temp. shift (ΔT_0 vs ΔT_{41J})

Background

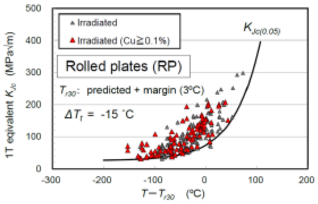
- Both JEAC4206-2007 and the newly proposed JEAC4206-2016 postulate that the temperature shifts of the reference (ΔT_0) and transition (ΔT_{41J}) temperatures are identical for determining the fracture toughness during the evaluated period.



JEAC 4206-2007

$K_{Ic} = 20.16 +129.9exp[0.0161(T-T_p)]$

- Lower bound curve



JEAC 4206-2016

$K_{Jc} = 25.2 + 36.6 exp[0.019\{T-(T_{r30} +\Delta T_t)\}]$

- Use of Master Curve concept

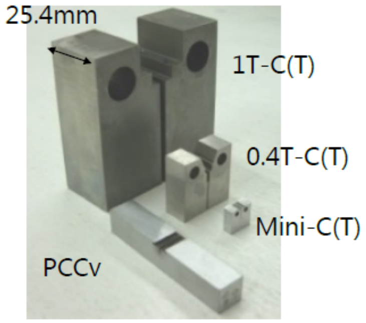
Objective

- To confirm the relationship between ΔT_{41J} and ΔT_0 at high fluence region

Size dependency of fracture toughness

Background

- Using T_0 is allowed to determine a fracture toughness curve in the newly proposed code (JEAC4206-2016).
- To evaluate T_0, it is necessary to conduct the fracture toughness tests using Mini-C(T) cut-off from the surveillance specimen.



Objective

- To ensure that T_0 is adequately evaluated by Mini-C(T) compared with that by 1T-C(T)

Prediction of irradiation embrittlement

-Statistical analysis of surveillance data and microscopic analysis-

Background

- The prediction equation of irradiation embrittlement of RPV steels was revised in JEAC4201-2007 (2013 addendum) reflecting the accumulation of surveillance data in a high fluence region.

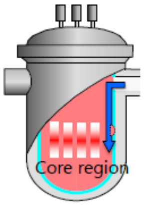
Objectives

- To confirm the conservatism of the prediction model, through the statistical analysis of surveillance data
- To confirm the adequacy of the prediction model in a high fluence region by obtaining microscopic observation data such as clusters of solute atoms and grain boundary segregation
- To identify major parameters potential for improving the prediction accuracy from the aforementioned results

Probabilistic Fracture Mechanics (PFM)

Work items

- Improvement of the functions of PFM analysis code "PASCAL" (JFY2012-2018)
- Development of the PFM guidebook (JFY2012-2017)
- Benchmark analysis with the other PFM analysis code (JFY2016-2017)



PASCAL

PFM analysis code for RPV considering neutron irradiation embrittlement & pressurized thermal shock (PTS) events developed by Japan Atomic Energy Agency (JAEA)

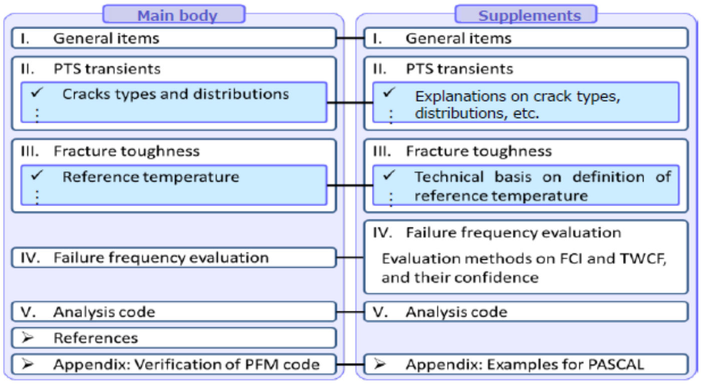
Improvement of the functions of PASCAL

In order to improve the accuracy of PFM analyses, following functions of PASCAL has been improved.

- Irradiation embrittlement prediction method
- Fracture toughness
- Loading conditions under PTS events
- Stress intensity factor K_I calculation
- Evaluation model for Non-Destructive Examination (NDE)
- Evaluation model for warm pre-stress effect

Development of the PFM guidebook

The PFM guidebook for RPVs has been developed to support calculations of probabilistic numerical index such as through-wall crack frequency (TWCF) by PFM analyses.



Main body	Supplements
I. General items	I. General items
II. PTS transients ✓ Cracks types and distributions ⋮	II. PTS transients ✓ Explanations on crack types, distributions, etc. ⋮
III. Fracture toughness ✓ Reference temperature ⋮	III. Fracture toughness ✓ Technical basis on definition of reference temperature ⋮
IV. Failure frequency evaluation	IV. Failure frequency evaluation Evaluation methods on FCI and TWCF, and their confidence
V. Analysis code	V. Analysis code
➢ References	
➢ Appendix: Verification of PFM code	➢ Appendix: Examples for PASCAL

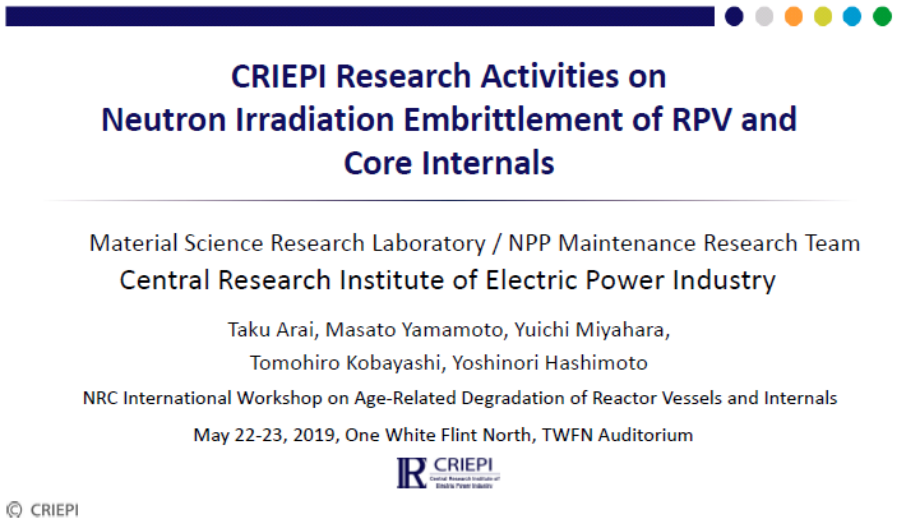
Reference: Katsuyama, J. et al., Guideline on Probabilistic Fracture Mechanics Analysis for Japanese Reactor Pressure Vessels, PVP2017-65921

Benchmark Analysis

- Benchmark analysis was conducted between PASCAL and FAVOR which has been utilized in the US regulation.
- Same Input data (pressure and temperature transitions, material properties, crack shapes, etc.) were used, and then different analysis models of the two codes were replaced with the same models step by step in the analyses.
- The following was confirmed:
  - the difference of K_I solutions was the most effective factor in the difference of the TWCF obtained from the two codes and
  - the results obtained from the two codes showed good agreement when the same analysis models were used.

Thank you for your attention.

B.6 CRIEPI Research Activities on Neutron Irradiation Embrittlement of RPV and Core Internals (T. Arai)

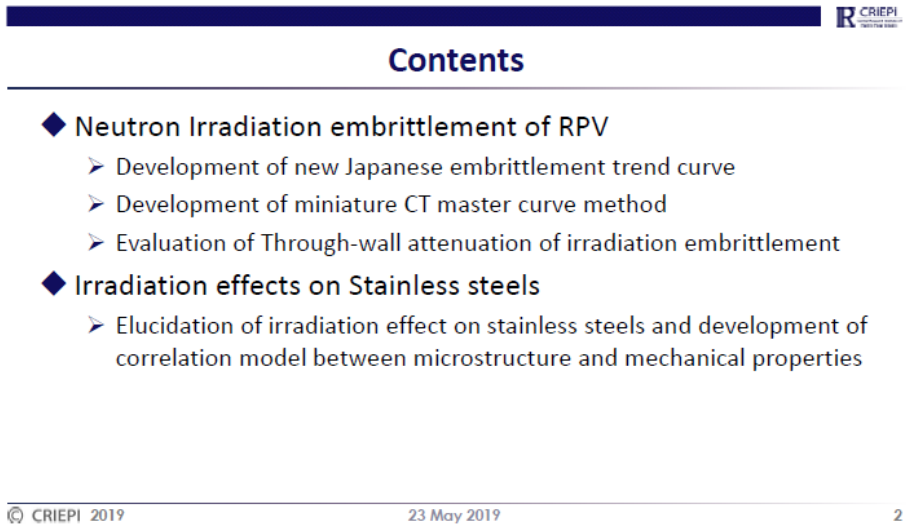


CRIEPI Research Activities on Neutron Irradiation Embrittlement of RPV and Core Internals

Material Science Research Laboratory / NPP Maintenance Research Team
Central Research Institute of Electric Power Industry

Taku Arai, Masato Yamamoto, Yuichi Miyahara,
Tomohiro Kobayashi, Yoshinori Hashimoto

NRC International Workshop on Age-Related Degradation of Reactor Vessels and Internals
May 22-23, 2019, One White Flint North, TWFN Auditorium



Contents

- Neutron Irradiation embrittlement of RPV
  - Development of new Japanese embrittlement trend curve
  - Development of miniature CT master curve method
  - Evaluation of Through-wall attenuation of irradiation embrittlement
- Irradiation effects on Stainless steels
  - Elucidation of irradiation effect on stainless steels and development of correlation model between microstructure and mechanical properties

Development of new Japanese embrittlement trend curve(ETC)

- JEAC4201-2007 adopted the CRIEPI's model which is guided by the understanding on the mechanism of embrittlement
  - Atom Probe Tomography (APT) data were very limited
- JEAC4201-2013 is a revised ETC of JEAC4201-2007
  - Baseline model remained the same
  - Coefficients were re-calibrated using the high-fluence surveillance data
- Current situation
  - No new surveillance data have been obtained over the last couple of years.
  - A large mount of microstructural data, particularly the Atom Probe Tomography (APT) data, have been accumulated up to now.
  - CRIEPI started developing new ECT to revise the current JEAC4201-2013 ETC from 2016.

JEAC 4201, Method of Surveillance Tests for Structural Materials of Nuclear Reactors , Japan Electric Association,

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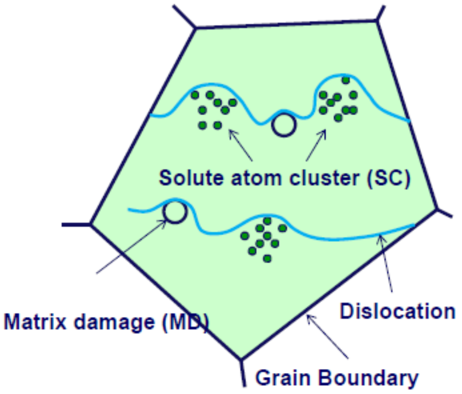
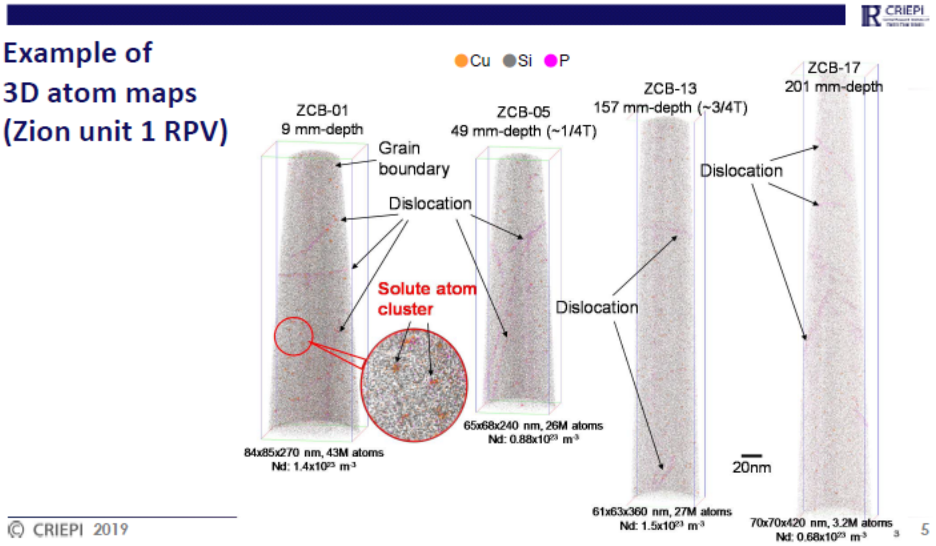
Basic idea of microstructural changes in RPV steel



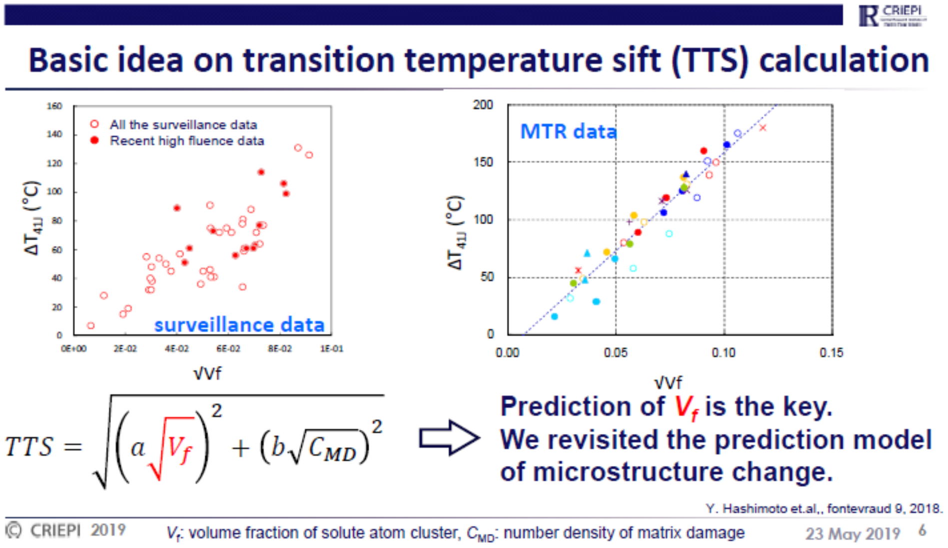
KEY PLAYERS

- Solute atom cluster
  - Irradiation enhanced cluster (homogeneous nucleation)
  - Irradiation induced cluster (heterogeneous nucleation)

  → detected by APT
- Matrix damage
  - Naked dislocation loops?



Example of 3D atom maps (Zion unit 1 RPV)



Basic idea on transition temperature sift (TTS) calculation

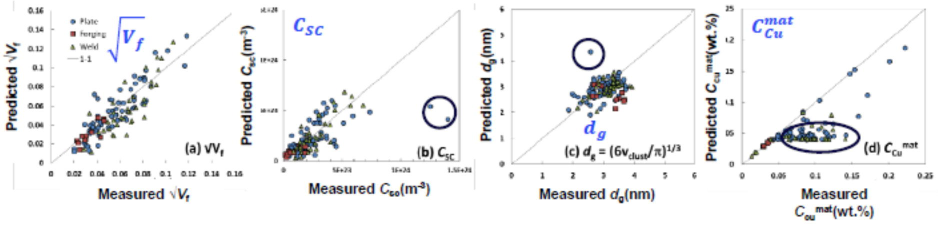
$$TTS = \sqrt{\left(a\sqrt{V_f}\right)^2 + \left(b\sqrt{C_{MD}}\right)^2} \Rightarrow$$

Prediction of V_f is the key.
We revisited the prediction model of microstructure change.

Y. Hashimoto et.al,, fontevraud 9, 2018.

V_f: volume fraction of solute atom cluster, C_{MD}: number density of matrix damage

JEAC4201-2013 prediction of microstructure change



The V_f prediction is in good agreement with the APT data.
The trends of predictions of C_{sc}, d_g and matrix Cu are roughly similar to the APT data.
There is room for improvement of prediction performance

⬇

Generalized form, Consideration of other elements, Fitting to the APT data and No use of differential equations

V_f: volume fraction of solute atom cluster, C_{SC}: number density of solute atom cluster,
d_g: bulk Cu content, C_{Cu}^{mat}: Cu content in matrix

Y. Hashimoto et.al,, fontevraud 9, 2018.

Analytical solution of revised model

Once the chemical composition, temperature and flux are given, the variables are functions of time as follows:

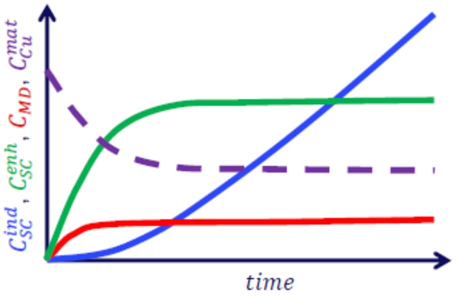
$$C_{SC}^{ind} = \gamma_1 t - \frac{\gamma_1}{\alpha_1}(1 - \exp(-\alpha_1 t))$$

$$C_{SC}^{enh} = \beta_4\left(1 - \frac{1}{\beta_3\beta_4 t + 1}\right)$$

$$C_{MD} = \frac{\gamma_1}{\alpha_1}(1 - \exp(-\alpha_1 t))$$

$$C_{Cu}^{mat} = C_{Cu}^{0} - \delta_1 C_{SC}^{enh}$$

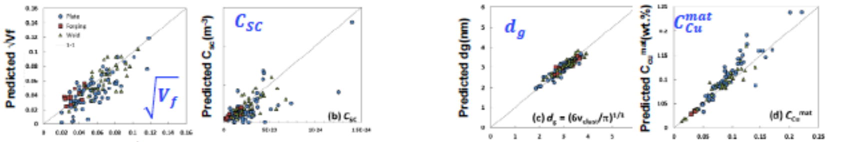
$$V_f = 1.1 v_{clust}(C_{SC}^{ind} + C_{SC}^{enh})$$



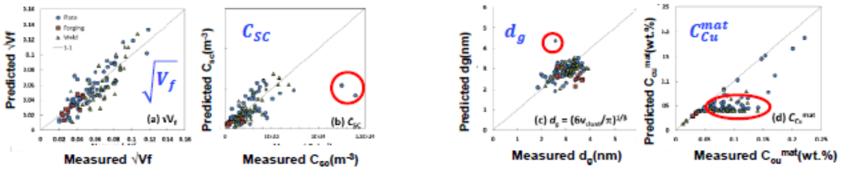
C_{MD}: number density of matrix damage
C_{SC}^{ind}: number density of irradiation induced solute atom cluster
C_{SC}^{enh}: number density of irradiation enhanced solute atom cluster
C_{Cu}^{mat}: Cu content in matrix
v_{Clust}: average cluster volume

Prediction of microstructure changes

Analytical solution $N_{clust}^{meas} \geqq 50$



JEAC4201-2013

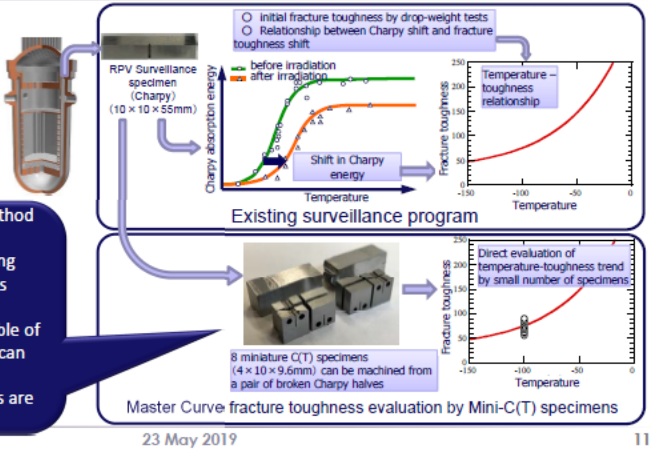


- The draft revised ETC model have been reviewed by experts of academic society in Japan.
- We would like to propose new ETC model to Japan Electric Association as soon as the review is over.
- We expect that new ETC can be implemented into JEAC 4201 in 2019 or 2020 JPY.

Y. Hashimoto et.al,, fontevraud 9, 2018.

Development of miniature C(T) master curve method

The Master Curve + Miniature C(T) specimens



○ initial fracture toughness by drop-weight tests
○ Relationship between Charpy shift and fracture toughness shift

Shift in Charpy energy

Temperature – toughness relationship

Existing surveillance program

8 miniature C(T) specimens (4 × 10 × 9.6mm) can be machined from a pair of broken Charpy halves

Direct evaluation of temperature-toughness trend by small number of specimens

Master Curve fracture toughness evaluation by Mini-C(T) specimens

The **Master Curve** method is a method to **directly determine fracture toughness** by statistically processing data from small number specimens based on Weibull distribution.
The **Master Curve** method is capable of the **specimen size correction.** We can use very small specimens if the appropriateness of test techniques are confirmed

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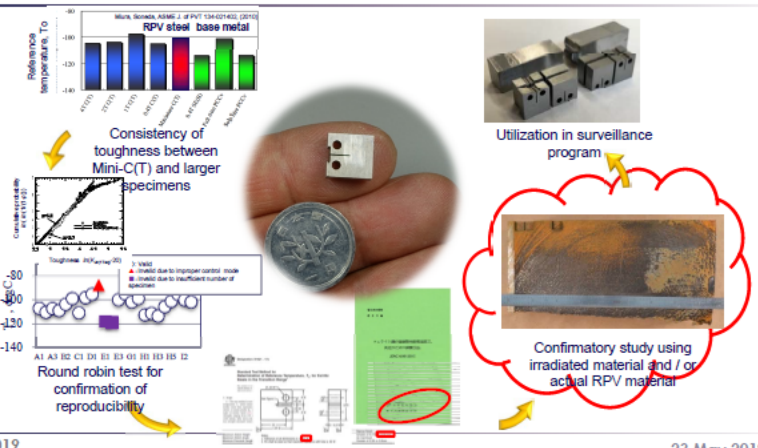
Development of Mini-C(T) technique



Consistency of toughness between Mini-C(T) and larger specimens

Round robin test for confirmation of reproducibility

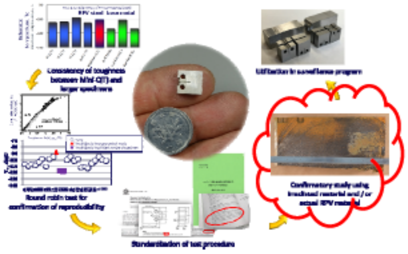
Standardization of test procedure

Utilization in surveillance program

Confirmatory study using irradiated material and / or actual RPV material

Step by step results

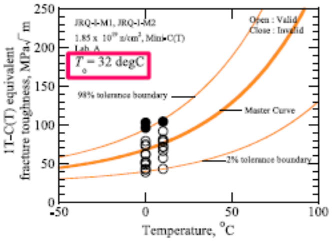
ONGOING CONFIRMATORY STUDY USING IRRADIATED MATERIALS



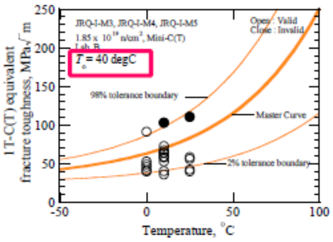
Irradiated material

- ✓ IAEA JRQ plate
- ✓ Linde 80 weld metal
- ✓ Zion Unit RPV base metal

Mini-C(T) application on IAEA irradiated JRQ



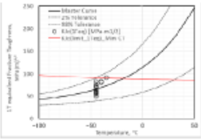
Irradiated Mini-C(T) by Lab. A

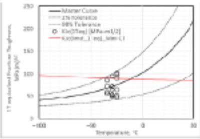


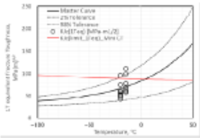
Irradiated Mini-C(T) by Lab. B

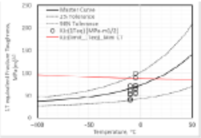
- ➢ T_o could be evaluated in both of Laboratories A and B on irradiated JRQ.
- ✓ T_o from Labs. A and B (32ºC, 40ºC) is consistent with each other.
- ✓ 5 of 25 (20%) Mini-C(T) specimens from Lab. A are out of 2 and 98 % tolerance boundary.
- ✓ 3 of 24 (13%) Mini-C(T) specimens from Lab. B are out of 2 and 98 % tolerance boundary.

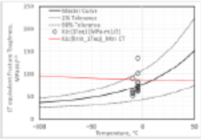
Mini-C(T) application on Zion unit 1 (decommissioned PWR) RPV plate material

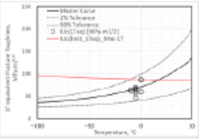

Layer 1, T-L, T_o= -9.7ºC

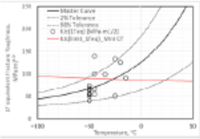

Layer 2, T-L, T_o= -2.2ºC

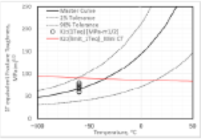

Layer 3, T-L, T_o=14.5ºC


Layer 4, T-L, T_o=25.5ºC

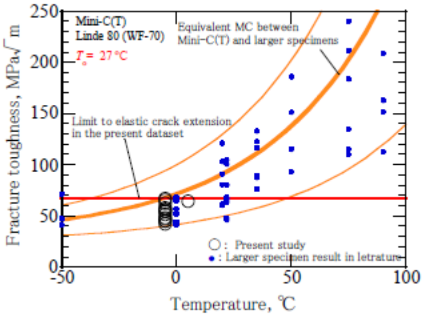

Layer 5, T-L, T_o=20.7ºC


Layer 7, T-L, T_o=28.8ºC


Layer 16, T-L, T_o= -16.8ºC


Layer 17, T-L, T_o= -25.7ºC

Mini-C(T) application on low upper shelf weld metal, Linde 80, in irradiated state



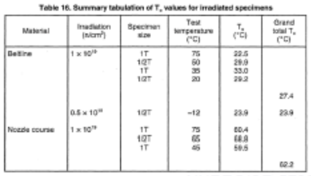


Table 16. Summary tabulation of T_o values for irradiated specimens

Material	Irradiation (n/cm²)	Specimen size	Test temperature (°C)	T_o (°C)	Grand total T_o (°C)
Beltline	1×10^{19}	1T	75	22.5	
		1/2T	50	29.9	
		1T	35	33.0	
		1/2T	20	29.2	
					27.4
	0.5×10^{19}	1/2T	−12	23.9	23.9
Nozzle course	1×10^{19}	1T	75	60.4	
		1/2T	65	68.8	
		1T	45	59.5	
					62.2

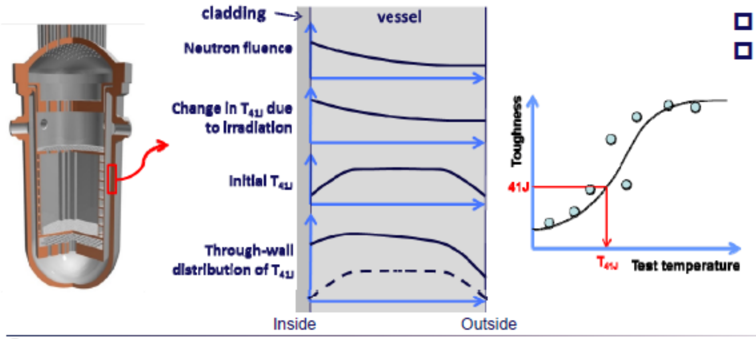
NUREG/CR-5736

- Linde 80 weld metal retrieved from unused Midland NPP RPV beltline weld
- Fluence (Irradiated in HSST project): 1 × 10^{19} n/cm² (>1MeV), 288℃
- Excellent agreement between Mini-C(T) and larger specimens

M. Yamamoto, ASME PVP2017-84906

Evaluation of Through-wall attenuation of irradiation embrittlement

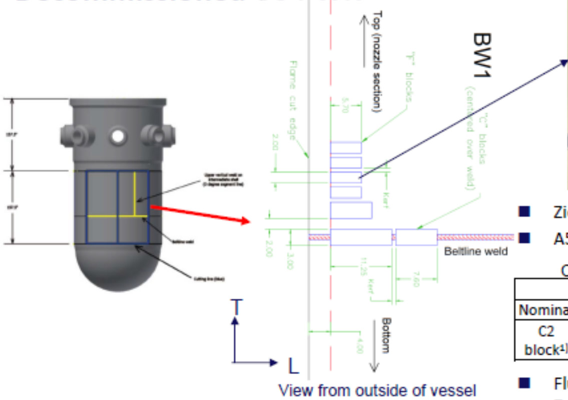
- Investigate combined effect of initial toughness distribution and fluence attenuation in a RPV steel to identify the conservatism in integrity assessment.

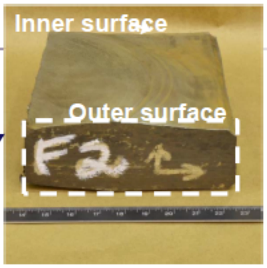


☐ IAEA JRQ
☐ Zion unit 1 RPV

↑ Collaboration with LWRS by ORNL

Zion unit 1 martial
Decommissioned US PWR





- Zion Unit 1 block F02 (L216xT73xS216 mm)
- A533B plate base metal

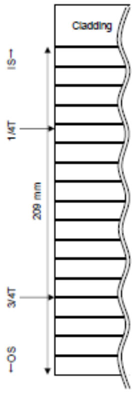
Chemical composition of Zion RPV base metal (wt%)

	Cu	Ni	Mn	Si	P
Nominal	0.11	0.49	1.30	-	0.001
C2 block[1)]	0.11–0.15	0.44–0.58	1.26–1.52	0.20–0.30	0.006–0.016

- Fluence at clad – base metal interface (CBMI) : $7x10^{18}$ n/cm^2(E > 1MeV)

1) ORNL/TM-2018/861

Investigation on Zion Unit 1 material



Layer No.	ID	Sampling depth (mm)	Fluence ($\times 10^{18}$ n/cm^2)	Flux ($\times 10^{10}$ n/cm^2·s)	APT	Vickers hardness	Metallographic	MC(T)_T-L	MC(T)_L-S	EPMA
1	ZCB-01	9	6.4	1.4	✓	✓	✓	✓	✓	✓
2	ZCB-02	19	5.8	1.2	✓	✓	✓	✓	✓	✓
3	ZCB-03	29	5.3	1.1	✓	✓	✓	✓	✓	
4	ZCB-04	39	4.8	1.0	✓	✓	✓	✓	✓	
5	ZCB-05	49	4.4	0.93	✓	✓	✓	✓	✓	✓
6	ZCB-06	59	4.0	0.85	–	–	–	–	–	–
7	ZCB-07	70	3.6	0.77	✓	✓	✓	✓	✓	✓
8	ZCB-08	93	2.9	0.61	–	–	–	–	–	–
9	ZCB-09	117	2.3	0.49	–	–	–	✓	–	–
10	ZCB-10	127	2.1	0.44	–	–	–	–	–	–
11	ZCB-11	138	1.9	0.40	–	–	–	✓	–	–
12	ZCB-12	147	1.7	0.37	–	–	–	–	–	–
13	ZCB-13	157	1.6	0.34	✓	✓	✓	✓	–	✓
14	ZCB-14	169	1.4	0.30	–	✓	–	✓	✓	–
15	ZCB-15	180	1.3	0.27	–	✓	–	✓	✓	–
16	ZCB-16	192	1.1	0.24	–	✓	–	✓	✓	–
17	ZCB-17	201	1.0	0.22	✓	✓	✓	✓	✓	✓

- Fluence is calculated using eq.(1)

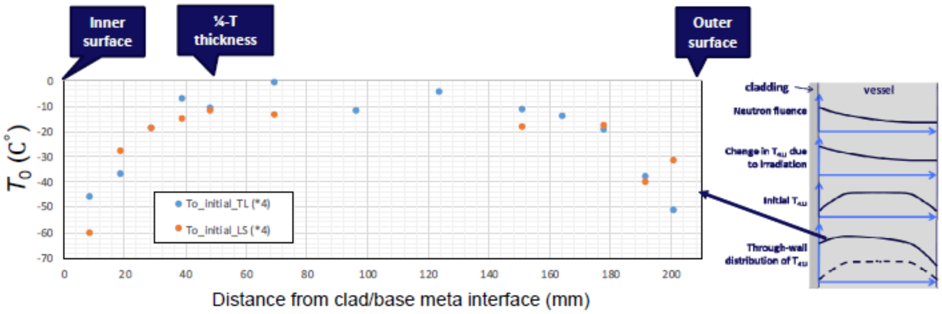
  $F = F_0 \cdot \exp(-0.24/25.4 \cdot a)$ (1)

  F_0: fluence at inner surface of RPV

  a: sampling depth (mm)

- Flux is calculated using fluence and 15 EFPY

Through-wall Fracture Toughness (T_o) distribution of Zion Unit 1 RPV



◆ Mini-C(T) technology confirmed the very high initial fracture toughness (low T_o) in inner and outer surface region than ¼-T location, where the surveillance specimens are machined.

Irradiation effect on stainless steels

Back ground

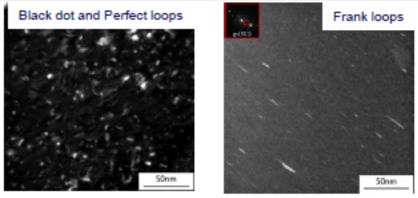
- The microstructural and mechanical properties of the internal components in a light water reactor core change under different operating conditions.
- Conventionally, microstructural evolution has been performed using TEM.
- Recently, trace element segregations at the grain boundaries and solute-enriched clusters in the matrices have been observed by atom probe tomography (APT).
- However, quantitative data supporting these findings are limited.

Objectives

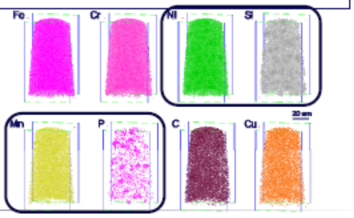
- Development of irradiated microstructure database, including APT as well as TEM.
- Clarify the correlation between microstructure and yield strength (hardness) to estimate the materials degradation(IASCC, fracture toughness).

Microstructural characterization by TEM and APT

Irradiation	Material	Project
Thermal reactor	316L (used in JNES IASCC project)	Japanese national project (sponsored by METI) "Improvement of Evaluation Methods of Irradiation Effects on Reactor Pressure Vessel and Core Internals" , collaborating with LWRS project
	304L (used in JNES IASCC project)	
Fast reactor	316 (used in CIR project)	
	304L (used in CIR project)	
PWR	347 (BFB from Westinghouse Two-loop down-flow type PWR)	
PWR (Chooz A)	304 (from decommissioned French PWR)	MAI-VIP

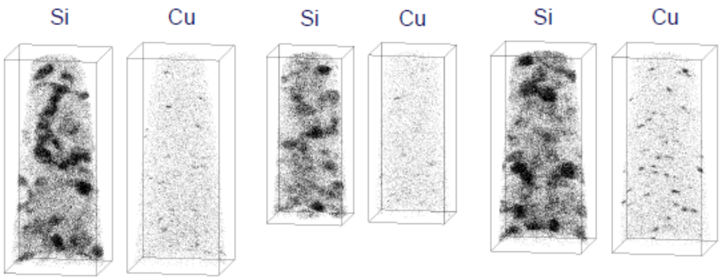


TEM (Black dot and Perfect loop, Frank loop)



APT (Solute enriched cluster)

APT Maps: Type 347

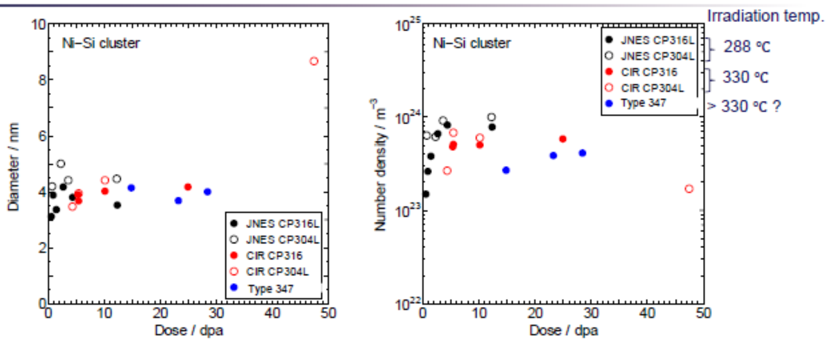


14.9 dpa
52 × 54 × 125 nm,
6.0 M atoms

23.3 dpa
42 × 44 × 98 nm,
3.4 M atoms

28.5 dpa
50 × 53 × 110 nm,
5.8 M atoms

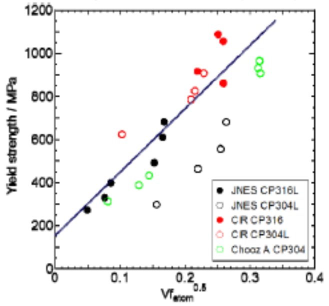
APT: Dose dependence of cluster size and number density



- Ni-Si cluster size rapidly reaches to 3∼5 nm then do not change dose up to 25 dpa.
- Ni-Si cluster size increase to 8.5 nm at 47.5 dpa
- Number density of Ni-Si cluster rapidly increase less than 1 dpa then tend to reach saturation level at 5 ∼10 dpa.
- Number density of Ni-Si cluster tends to be low when the irradiation temperature is high.

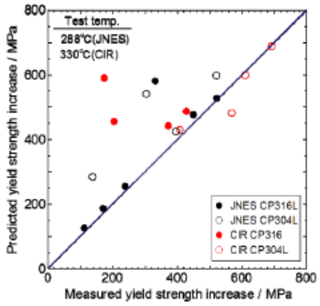
Microstructure and yield strength correlation

Empirical model (APT)



- ➢ The $Vf^{1/2}$ of clusters have a relatively good correlation with the yield strength.

Theoretical model (TEM and APT)



- ➢ Orowan based dispersed barrier model provide a good correlation with the yield strength increase.

Summary

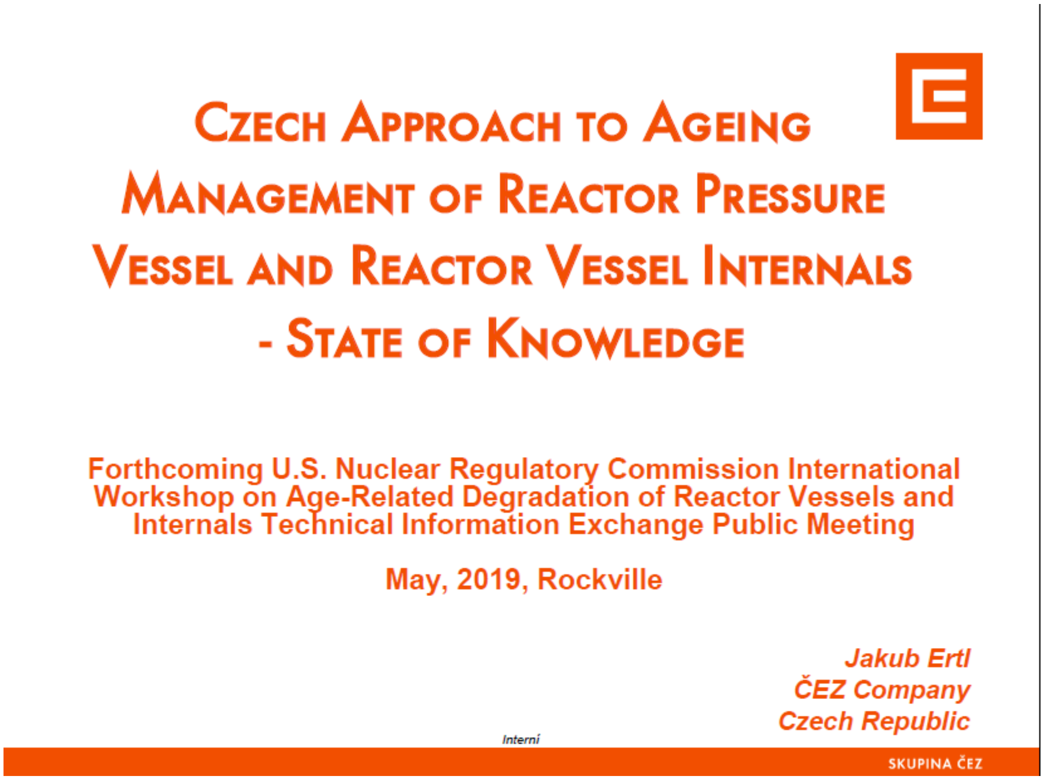
- ◆ Neutron Irradiation embrittlement of RPV
  - ➢ Based on the accumulated APT analysis data, the prediction accuracy of the microstructural change by irradiation was improved. We are examining a new ETC that incorporates this improvement.
  - ➢ After basic studies on the feasibility of the Mini-C(T) master curve technique, we are conducting research to confirm the applicability to irradiated materials.
  - ➢ We investigated through wall distribution of macro and micro-structure, hardness, toughness in Zion unit 1 PRV. We obtained data which shows combined effects of initial distribution and fluence attenuation in a RPV steel .
- ◆ Irradiation effects on Stainless steels
  - ➢ We conducted TEM and ATP analyses on irradiated stainless steels with various material and irradiation conditions and enhanced data base on microstructure of irradiated stainless steels.
  - ➢ We are developing the correlation model between microstructure and mechanical properties based on these findings.

Acknowledgment

- Following Research items of were conducted in the national project of "Improvement of Evaluation Methods of Irradiation Effects on Reactor Pressure Vessel and Core Internals" (2015 to 2018 JFY) sponsored by METI. The project was collaborated with LWRS project by ORNL.
  - Development of miniature CT master curve method (for Irradiated material)
  - Evaluation of Through-wall attenuation of irradiation embrittlement
  - Elucidation of irradiation effect on stainless steels and development of correlation model between microstructure and mechanical properties (except Chooz A material)

Supplemental slides

B.7 Czech Approach to Ageing Management of Reactor Pressure Vessel and Reactor Vessel Internals—State of Knowledge (J. Ertl)



CZECH APPROACH TO AGEING MANAGEMENT OF REACTOR PRESSURE VESSEL AND REACTOR VESSEL INTERNALS - STATE OF KNOWLEDGE

Forthcoming U.S. Nuclear Regulatory Commission International Workshop on Age-Related Degradation of Reactor Vessels and Internals Technical Information Exchange Public Meeting

May, 2019, Rockville

Jakub Ertl
ČEZ Company
Czech Republic

Interní

SKUPINA ČEZ

CONTENT

- BASIC INFORMATION ABOUT CZECH NPPs
- LTO DEPARTMENT
- AGEING MANAGEMENT PROGRAMS FOR CZECH REACTORS
- SURVEILLANCE PROGRAM IN DUKOVANY NPP
- SURVEILLANCE PROGRAM IN TEMELIN NPP
- METHODOLOGY OF RADIATION DAMAGE EVALUATION
- FAST NEUTRON FLUENCE EVALUATION
- AGEING MANAGEMNT PROGRAM FOR RPV INTERNALS

Interní

SKUPINA ČEZ

DESCRIPTION OF DUKOVANY NPP

4 units - WWER 440/213 type, six-*loop primary circuit* (second generation)

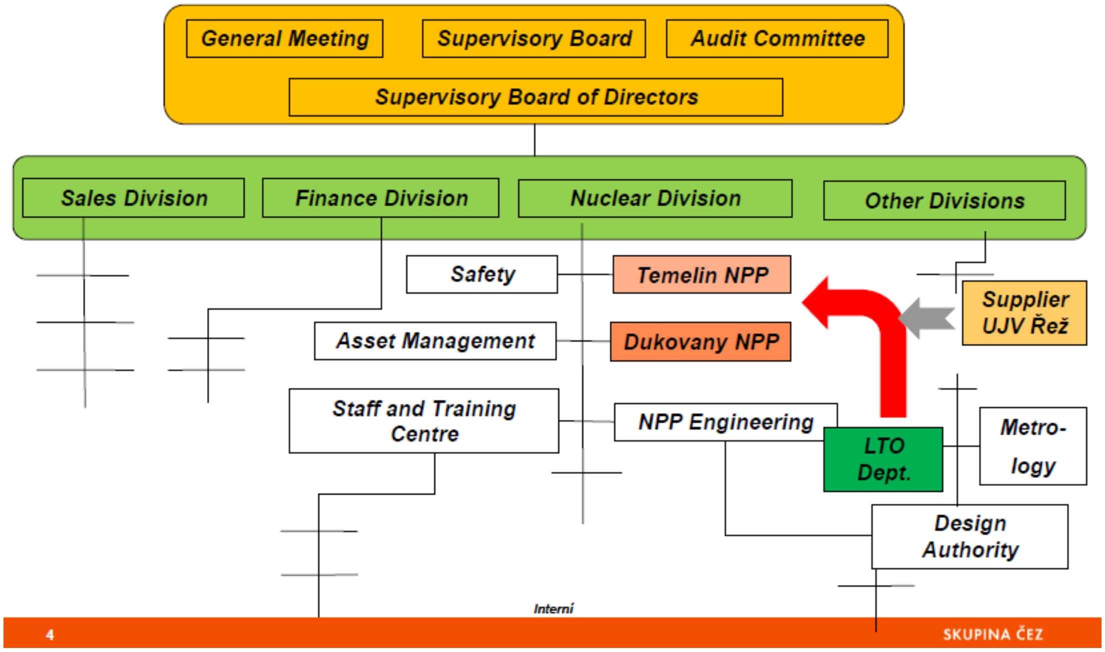
- **Water cooled, Water moderated, (Energy) Reactor – WWER,** (Pressurized Water Reactor – PWR)
- Each reactor loop has a horizontal steam generator, main circulation pump and main gate valve
- Unit 1 was put to operation in 1985 (Unit 2 and 3 – 1986, Unit 4 – 1987)
- Power up-rate 105 % to 510 MWe /unit within 2009-2012
- Design lifetime 30 years (only RPVs have 40 years), LTO prospect up to 60 years

DESCRIPTION OF TEMELIN NPP

- **Two Units WWER 1000/V320 type, four-loop primary circuit**
- Each reactor loop has a horizontal steam generator and main circulation pump
- Containment made of sealed reinforced concrete with the inner steel liner
- Triple redundancy of emergency reactor core cooling systems
- One 1000 MW turbine generator for each reactor Unit
- Unit 1 was put to Operation in 2000 (Unit 2 in 2002)

ORGANISATIONAL CHART OF ČEZ, A.S.

LTO DEPARTMENTT IS RESPONSIBLE FOR AMPS AND TLAAS



General Meeting

Supervisory Board

Audit Committee

Supervisory Board of Directors

Sales Division

Finance Division

Nuclear Division

Other Divisions

Safety

Temelin NPP

Asset Management

Dukovany NPP

Staff and Training Centre

NPP Engineering

LTO Dept.

Supplier UJV Řež

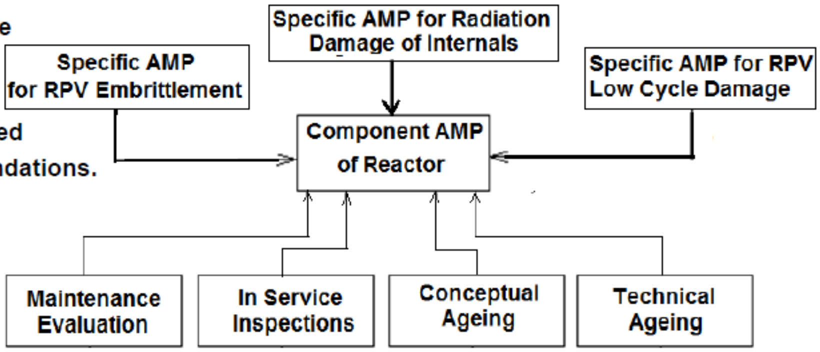
Metro-logy

Design Authority

Interní

AGEING MANAGEMENT PROGRAMMES (AMPS)

- Two types of AMPs
  - **Component specific AMP** – it is focused on components (reactor, steam generator, main circulation pump, main closing valve, …)
  - **Specific AMP** - it is focused on degradation mechanism (low-cycle fatigue, flow accelerated corrosion, radiation embrittlement, …)
- **Specific AMPs were created according to generic attributes prescribed by IAEA recommendations.**



Specific AMP for Radiation Damage of Internals

Specific AMP for RPV Embrittlement

Specific AMP for RPV Low Cycle Damage

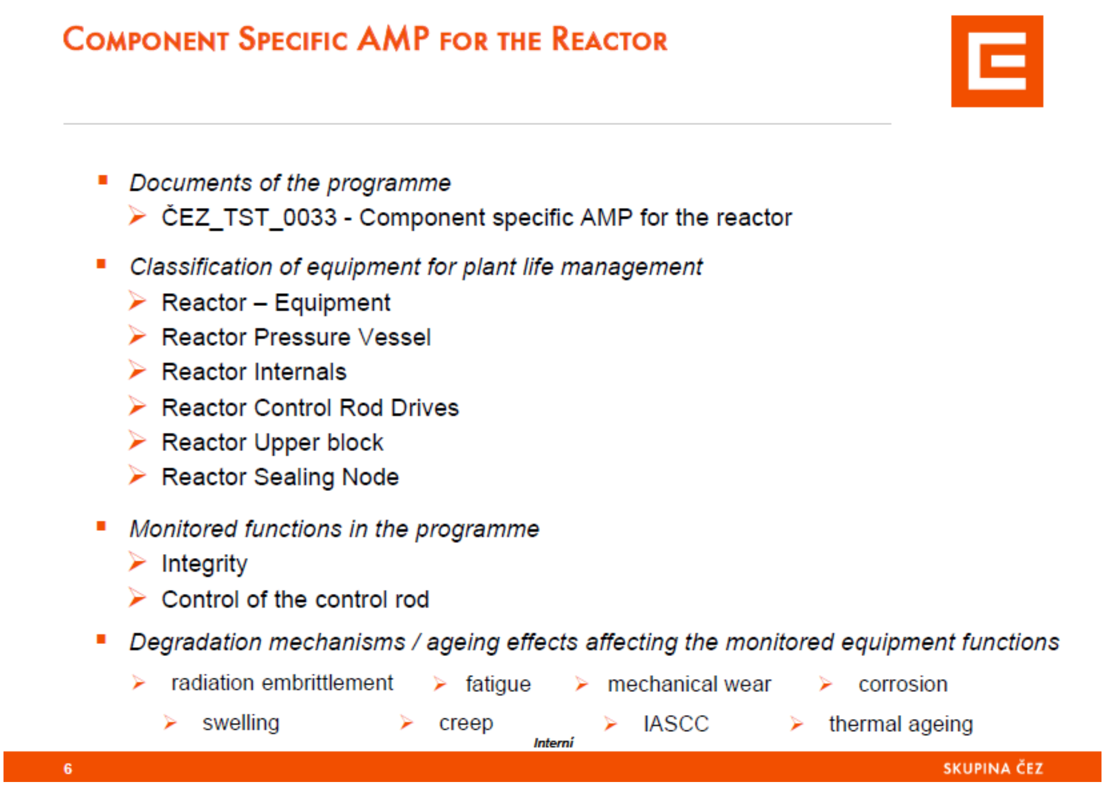
Component AMP of Reactor

Maintenance Evaluation

In Service Inspections

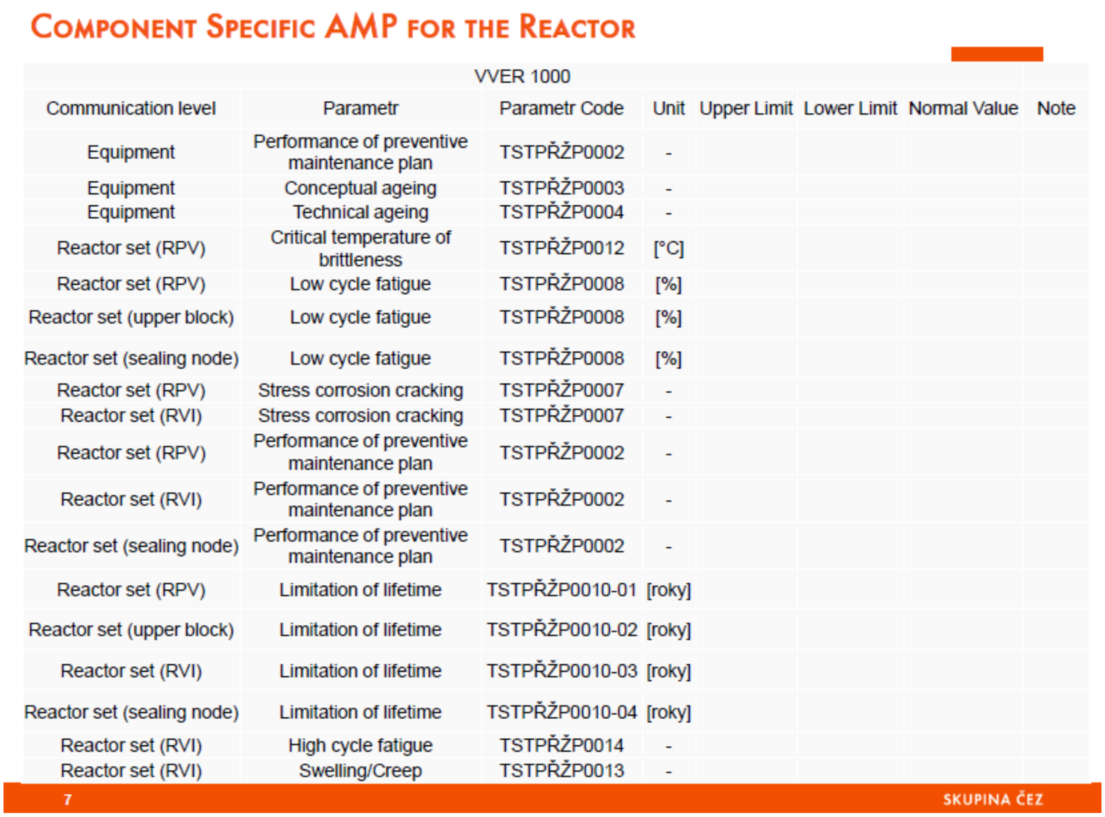
Conceptual Ageing

Technical Ageing



COMPONENT SPECIFIC AMP FOR THE REACTOR

- *Documents of the programme*
  - ČEZ_TST_0033 - Component specific AMP for the reactor
- *Classification of equipment for plant life management*
  - Reactor – Equipment
  - Reactor Pressure Vessel
  - Reactor Internals
  - Reactor Control Rod Drives
  - Reactor Upper block
  - Reactor Sealing Node
- *Monitored functions in the programme*
  - Integrity
  - Control of the control rod
- *Degradation mechanisms / ageing effects affecting the monitored equipment functions*
  - radiation embrittlement
  - fatigue
  - mechanical wear
  - corrosion
  - swelling
  - creep
  - IASCC
  - thermal ageing

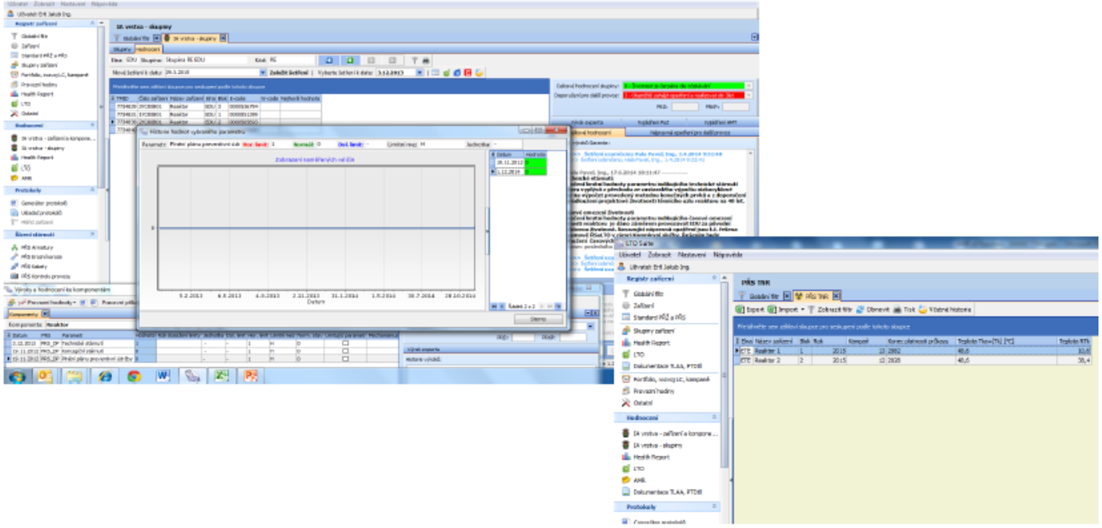


COMPONENT SPECIFIC AMP FOR THE REACTOR

		VVER 1000					
Communication level	Parametr	Parametr Code	Unit	Upper Limit	Lower Limit	Normal Value	Note
Equipment	Performance of preventive maintenance plan	TSTPŘŽP0002	-				
Equipment	Conceptual ageing	TSTPŘŽP0003	-				
Equipment	Technical ageing	TSTPŘŽP0004	-				
Reactor set (RPV)	Critical temperature of brittleness	TSTPŘŽP0012	[°C]				
Reactor set (RPV)	Low cycle fatigue	TSTPŘŽP0008	[%]				
Reactor set (upper block)	Low cycle fatigue	TSTPŘŽP0008	[%]				
Reactor set (sealing node)	Low cycle fatigue	TSTPŘŽP0008	[%]				
Reactor set (RPV)	Stress corrosion cracking	TSTPŘŽP0007	-				
Reactor set (RVI)	Stress corrosion cracking	TSTPŘŽP0007	-				
Reactor set (RPV)	Performance of preventive maintenance plan	TSTPŘŽP0002	-				
Reactor set (RVI)	Performance of preventive maintenance plan	TSTPŘŽP0002	-				
Reactor set (sealing node)	Performance of preventive maintenance plan	TSTPŘŽP0002	-				
Reactor set (RPV)	Limitation of lifetime	TSTPŘŽP0010-01	[roky]				
Reactor set (upper block)	Limitation of lifetime	TSTPŘŽP0010-02	[roky]				
Reactor set (RVI)	Limitation of lifetime	TSTPŘŽP0010-03	[roky]				
Reactor set (sealing node)	Limitation of lifetime	TSTPŘŽP0010-04	[roky]				
Reactor set (RVI)	High cycle fatigue	TSTPŘŽP0014	-				
Reactor set (RVI)	Swelling/Creep	TSTPŘŽP0013	-				

MAIN SOFTWARE USED FOR AGEING MANAGEMENT OF RPV AND LIFETIME ASSESSMENT OF REACTOR

- AMP outputs are stored in software LTOs and incorporated to **Health Reports, Safety Report, Periodic Lifetime Assessment**

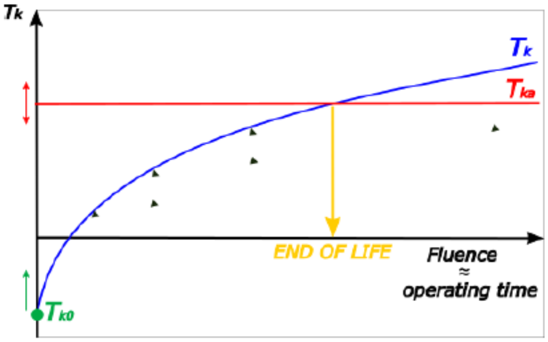


SPECIFIC AMP FOR RPV EMBRITTLEMENT

- **Documents**
  - ČEZ_ME_0780 Ageing Management Programme for RPV Embrittlement.
- The assessment of RPV embrittlement is performed in accordance with international methodology VERLIFE and Czech normative technical documentation A. M. E.
- The surveillance program is conducted by predefined harmonogram.

⬇

- We use surveillance program results for predictions of critical temperature of brittleness in the case of Dukovany NPP.

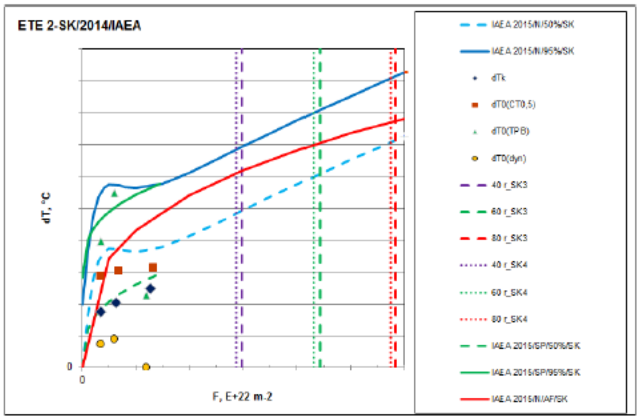


SPECIFIC AMP FOR RPV EMBRITTLEMENT

- The surveillance program of Temelin RPV is at the beginning stage.

- We are not able to use results of surveillance program for predictions of critical temperature of brittleness.

- **We are obliged to use normative curves for embrittlement assessment..**

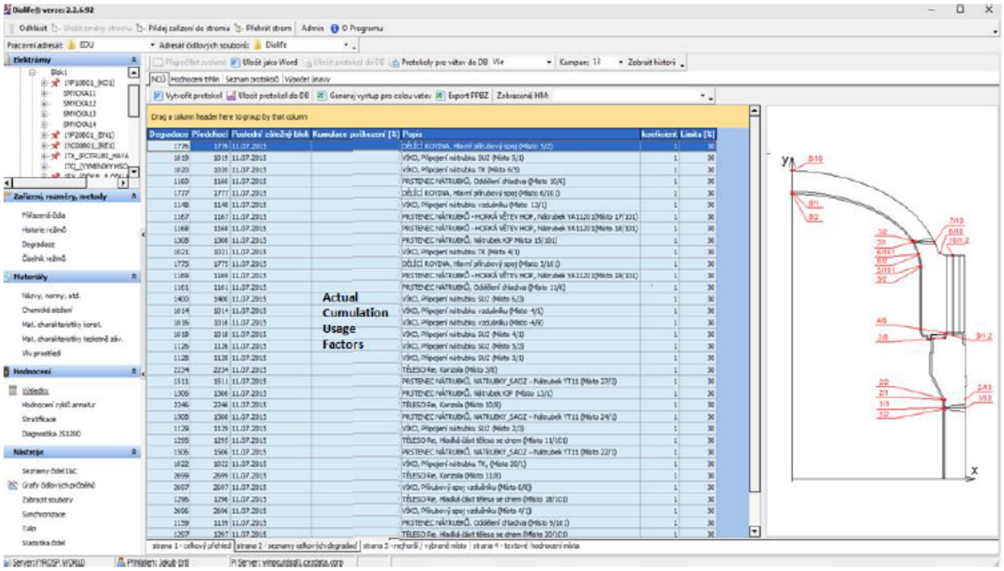


- New prediction formula for WWER-1000 RPV materials is based on results from the analysis of the database of surveillance specimen test results (after re-analysis of neutron fluence and reconstitution of specimens realized within European projects TACIS, TAREG and Russian projects)

SPECIFIC AMP FOR LOW – CYCLE FATIGUE OF RPV

- ***Documents***
  - *ČEZ_ME_0780 Ageing Management Programme for RPV Embrittlement.*
- **The Ageing Management Program for low-cycle fatigue consists of the following periodic activities:**
  - Collection of data necessary for evaluating fatigue of the equipment
  - Evaluation of the measured operating parameters
  - Calculation of the accumulation of fatigue (CUF)
  - Prediction of the residual fatigue lifetime
  - Proposal and implementation of corrective measures
  - SW DIALIFE is the executive tool of this AMP
  - The limit value for total fatigue is **100%.**
  - Acceptance criteria of further operation are specified (40%, 60% and 80%) and appropriate corrective actions are defined when every acceptance criterion is reached
  - AMP outputs are stored in software LTOs and incorporated to **Health Reports, Safety Report, Periodic Lifetime Assessment**

SPECIFIC AMP FOR LOW – CYCLE DAMAGE OF RPV



Dialife® verze 2.2.6.92

Odhlásit | Uložit změny stromu | Přidej zařízení do stromu | Přehrát strom | Admin | O Programu

Pracovní adresář: EDU | Adresář čidlových souborů: Dialife

Přepočítat zvolené | Uložit jako Word | Uložit protokol do DB | Protokoly pro větev do DB: Vše | Kampaň: 13 | Zobrait histori

NCÚ | Hodnocení trhlin | Seznam protokolů | Výpočet únavy

Vytvořit protokol | Uložit protokol do DB | Generuj výstup pro celou vetev | Export PPBZ | Zobrazené HM:

Drag a column header here to group by that column

Degradace	Předchozí	Poslední zátěžný blok	Kumulace poškození [%]	Popis	koeficient	Limita [%]
1776	1776	11.07.2015		DĚLÍCÍ ROVINA, Hlavní přírubový spoj (Místo 5/2)	1	30
1019	1019	11.07.2015		VÍKO, Připojení nátrubku SUZ (Místo 5/1)	1	30
1020	1020	11.07.2015		VÍKO, Připojení nátrubku TK (Místo 6/5)	1	30
1160	1160	11.07.2015		PRSTENEC NÁTRUBKŮ, Oddělení chladiva (Místo 10/6)	1	30
1777	1777	11.07.2015		DĚLÍCÍ ROVINA, Hlavní přírubový spoj (Místo 6/10 1)	1	30
1148	1148	11.07.2015		VÍKO, Připojení nátrubku vzdušníku (Místo 13/1)	1	30
1167	1167	11.07.2015		PRSTENEC NÁTRUBKŮ - HORKÁ VĚTEV HCP, Nátrubek YA11201(Místo 17/101)	1	30
1168	1168	11.07.2015		PRSTENEC NÁTRUBKŮ - HORKÁ VĚTEV HCP, Nátrubek YA11201(Místo 18/101)	1	30
1308	1308	11.07.2015		PRSTENEC NÁTRUBKŮ, Nátrubek KP Místo 15/101)	1	30
1021	1021	11.07.2015		VÍKO, Připojení nátrubku TK (Místo 4/1)	1	30
1775	1775	11.07.2015		DĚLÍCÍ ROVINA, Hlavní přírubový spoj (Místo 5/10 1)	1	30
1169	1169	11.07.2015		PRSTENEC NÁTRUBKŮ - HORKÁ VĚTEV HCP, Nátrubek YA11201(Místo 19/101)	1	30
1161	1161	11.07.2015		PRSTENEC NÁTRUBKŮ, Oddělení chladiva (Místo 11/6)	1	30
1400	1400	11.07.2015		VÍKO, Připojení nátrubku SUZ (Místo 6/3)	1	30
1014	1014	11.07.2015		VÍKO, Připojení nátrubku vzdušníku (Místo 4/1)	1	30
1016	1016	11.07.2015		VÍKO, Připojení nátrubku vzdušníku (Místo 4/6)	1	30
1018	1018	11.07.2015		VÍKO, Připojení nátrubku SUZ (Místo 4/1)	1	30
1126	1126	11.07.2015		VÍKO, Připojení nátrubku SUZ (Místo 5/3)	1	30
1128	1128	11.07.2015		VÍKO, Připojení nátrubku SUZ (Místo 3/1)	1	30
2234	2234	11.07.2015		TĚLESO Re, Konzola (Místo 3/8)	1	30
1511	1511	11.07.2015		PRSTENEC NÁTRUBKŮ, NATRUBKY_SAOZ - Nátrubek YT11 (Místo 27/3)	1	30
1306	1306	11.07.2015		PRSTENEC NÁTRUBKŮ, Nátrubek KP (Místo 13/1)	1	30
2246	2246	11.07.2015		TĚLESO Re, Konzola (Místo 10/8)	1	30
1508	1508	11.07.2015		PRSTENEC NÁTRUBKŮ, NATRUBKY_SAOZ - Nátrubek YT11 (Místo 24/1)	1	30
1129	1129	11.07.2015		VÍKO, Připojení nátrubku SUZ (Místo 2/3)	1	30
1295	1295	11.07.2015		TĚLESO Re, Hladká část tělesa se dnem (Místo 11/101)	1	30
1506	1506	11.07.2015		PRSTENEC NÁTRUBKŮ, NATRUBKY_SAOZ - Nátrubek YT11 (Místo 22/1)	1	30
1022	1022	11.07.2015		VÍKO, Připojení nátrubku TK, (Místo 20/1)	1	30
2699	2699	11.07.2015		TĚLESO Re, Konzola (Místo 11/8)	1	30
2697	2697	11.07.2015		VÍKO, Přírubový spoj vzdušníku (Místo 6/8)	1	30
1298	1298	11.07.2015		TĚLESO Re, Hladká část tělesa se dnem (Místo 18/101)	1	30
2696	2696	11.07.2015		VÍKO, Přírubový spoj vzdušníku (Místo 4/1)	1	30
1159	1159	11.07.2015		PRSTENEC NÁTRUBKŮ, Oddělení chladiva (Místo 9/10 1)	1	30
1297	1297	11.07.2015		TĚLESO Re, Hladká část tělesa se dnem (Místo 20/101)	1	30

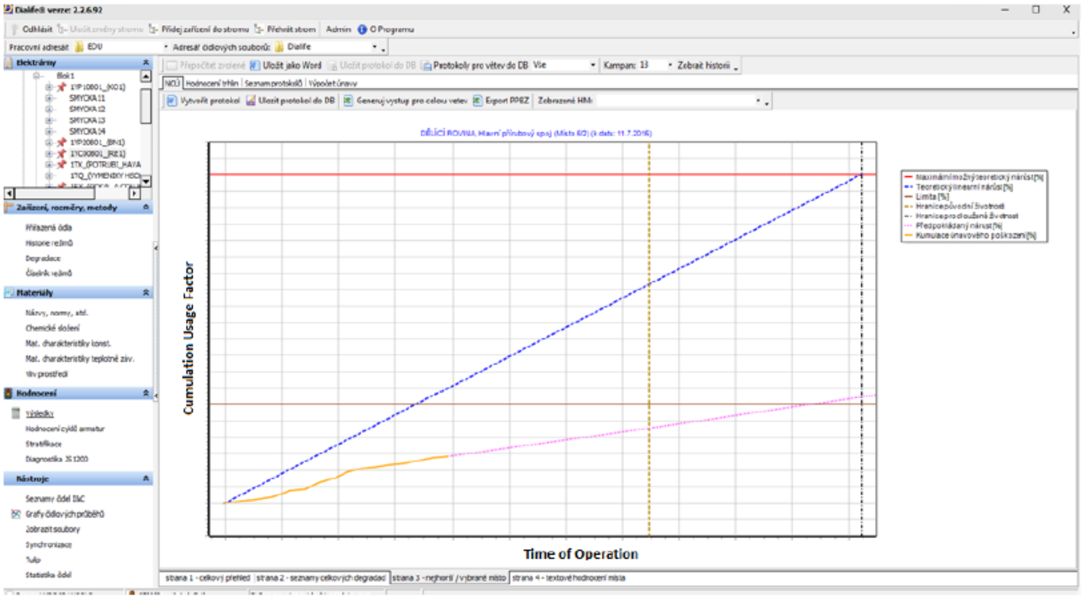
Actual Cumulation Usage Factors

strana 1 - celkový přehled | strana 2 - seznamy celkových degradací | strana 3 - nejhorší / vybrané místo | strana 4 - textové hodnocení místa

Přihlášen: Jakub Ertl | PI Server: vmpcudsql1.cezdata.corp

Interní

SPECIFIC AMP FOR LOW – CYCLE DAMAGE OF RPV



Dialife® verze 2.2.6.92

Odhlásit | Uložit změny stromu | Přidej zařízení do stromu | Přehrát strom | Admin | O Programu

Pracovní adresář: EDU | Adresář čidlových souborů: Dialife

Přepočítat zvolené | Uložit jako Word | Uložit protokol do DB | Protokoly pro větev do DB: Vše | Kampaň: 13 | Zobrait histori

NCÚ | Hodnocení trhlin | Seznam protokolů | Výpočet únavy

Vytvořit protokol | Uložit protokol do DB | Generuj výstup pro celou vetev | Export PPBZ | Zobrazené HM:

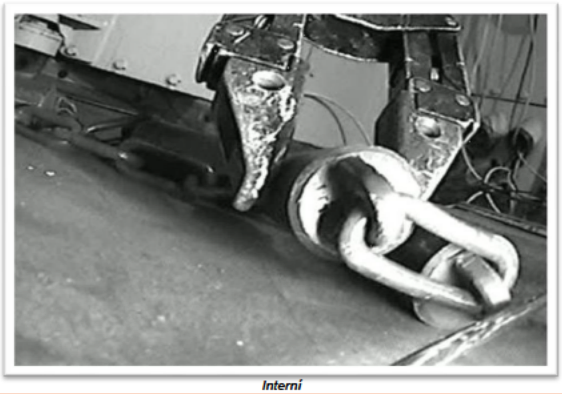
DĚLÍCÍ ROVINA, Hlavní přírubový spoj (Místo 8/2) (k datu: 11.7.2016)

strana 1 - celkový přehled | strana 2 - seznamy celkových degradací | strana 3 - nejhorší / vybrané místo | strana 4 - textové hodnocení místa

Interní

SURVEILLANCE PROGRAMS IN DUKOVANY NPP

- STANDARD SURVEILLANCE PROGRAM
- SUPPLEMENTARY SURVEILLANCE PROGRAM
- EXTENDED SURVEILLANCE PROGRAM



STANDARD SURVEILLANCE PROGRAM

Designed:	before beginning of operation
Monitored materials:	basic material, weld metal, heat affected zone
Types of tests:	Charpy test Fracture toughness test Tensile test

Program conception:

Monitoring of RPV material properties changes depending on operating time (40 years) with respect to:

- ✓ radiation embrittlement,
- ✓ thermal ageing,
- ✓ determination of irradiated materials of RPV after regenerative annealing.

STANDARD SURVEILLANCE PROGRAM

It was designed by Škoda JS at the knowledge level of 1970s. It was obvious (in 1990s during program evaluation) that the program is not enough for evaluation of status and prediction of residual lifetime of RPV.

The main shortcomings were:

- ✓ Temporality of program (5 years) – the approach was adequate to the knowledge of 1970s. A new approach was needed for a long term operation (to demonstrate knowledge of the state and to determine residual lifetime).
- ✓ Acceleration coefficient of surveillance specimens irradiation was too high.
- ✓ Impossibility to evaluate the irradiation temperature.
- ✓ Large uncertainty in fluence determination (insufficient number of monitors + unknown orientation of tested specimens against core).
- ✓ Absence of welding material.

SUPPLEMENTARY SURVEILLANCE PROGRAM

Start:	in 1997 at 4th block. Inserting and withdrawing according to the schedule, ending in 2023
Monitored materials:	basic material, weld metal, cladding, reference material
Types of tests:	Charpy test Fracture toughness test Tensile test

Program conception:

Monitoring of RPV material properties changes depending on operating time (40 years) with respect to:

- ✓ radiation embrittlement,
- ✓ radiation embrittlement after regenerative annealing,
- ✓ monitoring of irradiation conditions

It was designed by ÚJV Řež in cooperation with Škoda JS – shortcomings of standard surveillance program were removed.

EXTENDED SURVEILLANCE PROGRAM

Start: in 2010, inserting and withdrawing according to the schedule, ending in 2050

Monitored materials: basic material, weld metal, cladding, reference material, heat affected zone, welding repair material

Types of tests:
Charpy test
Fracture toughness test
Tensile test

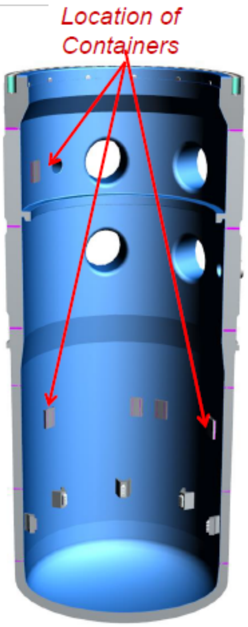
Program conception:
Monitoring of RPV material properties changes depending on operating time (up to 80 years) with respect to:

- ✓ radiation embrittlement,
- ✓ radiation embrittlement after regenerative annealing,
- ✓ monitoring of irradiation conditions.

SURVEILLANCE PROGRAM IN TEMELIN NPP

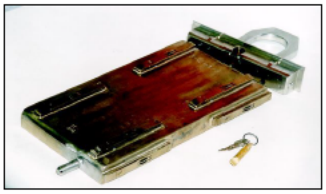
- **Six** containers are determined for the effect of radiation embrittlement, **two** for the effect of annealing and for the re-embrittlement effects, if necessary - all containers are located on inner RPV surface in the beltline region
- **Two** containers are determined for the effect of thermal aging – the containers are located above the core
- **Design of containers and holders allow withdrawal and also re-loading of new containers.**
- Three containers determined for the effect of radiation embrittlement were withdrawn from each unit

UNIT 1	UNIT 2
2004	2005
2008	2009
2012	2013

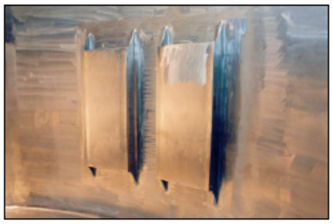


SURVEILLANCE PROGRAMS IN TEMELIN NPP

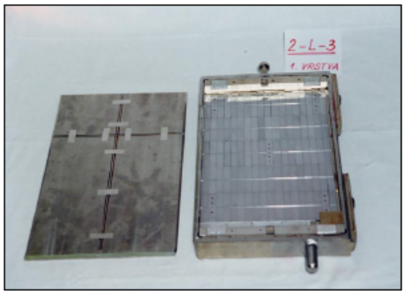
FLAT TYPE CONTAINER
200 X 300 X 25 mm



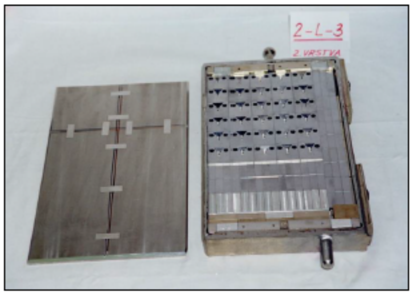
LOCATION OF CONTAINERS ON RPV WALL



First layer



Second layer



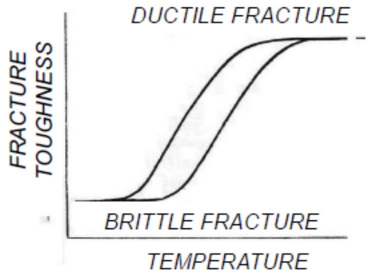
SURVEILLANCE PROGRAMS IN TEMELIN NPP

SPECIMENS TYPES

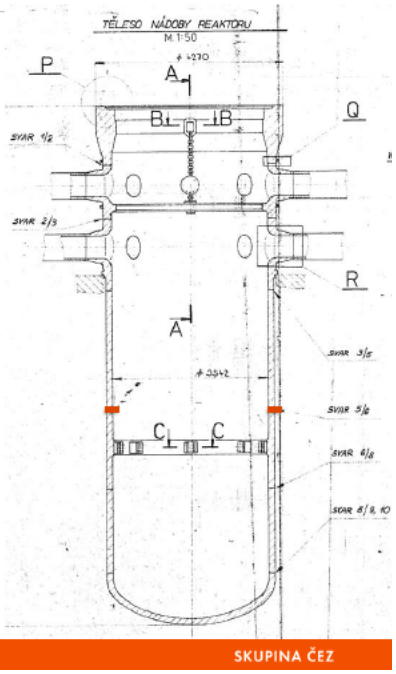
- **TENSILE SPECIMENS –** base metal and weld metal
- **SPECIMENS FOR IMPACT NOTCH TOUGHNESS -** base metal (24), weld metal (24), heat affected zone (24), JRQ (32) and materials from other WWER-1000 units (12 per material);
- **SPECIMENS FOR STATIC FRACTURE TOUGHNES OF CT 0.5 TYPE -** base metal (15) and weld metal(15);
- **SPECIMENS OF „COD“ (TPB) TYPE –** base metal (14), weld metal (14), heat affected zone(14), first and second layer of austenitic cladding (14), JRQ (14), and materials from other WWER-1000 units (12 per material);

EVALUATION PROCESS

- Radiation damage of basic material in the location of maximum fluence and radiation damage of 5/6 weld is based on shifts of transient temperature



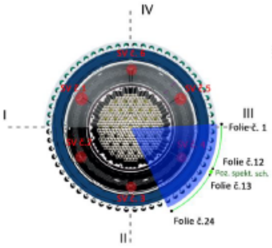
- Evaluation inputs:
  - fast neutron fluence,
  - transient temperature limit value
- Two types of evaluations:
  - **Shift of critical brittleness temperature**
  - **Determination of reference temperature**

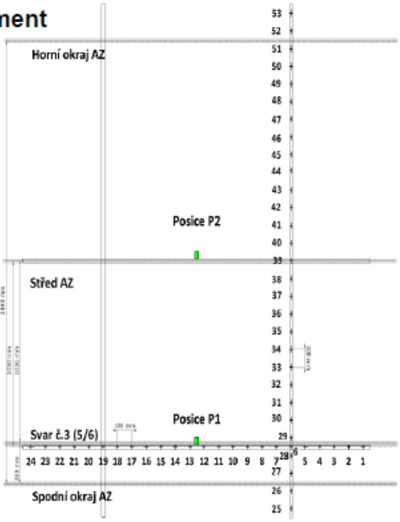


Interní

FLUENCE

- **RPV fluence is determined by calculation corrected by measurement in surveillance program and by ex-vessel measurement**
- Inserting/withdrawing of rack for ex-vessel measurement is done manually through inspection hole for NDT under reactor
- Rack is situated behind surveillance specimens channel with dosimetric chain

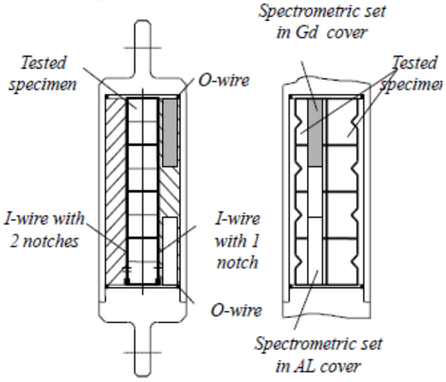


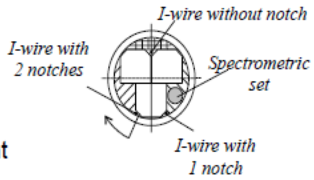


Interní

FLUENCE – SURVEILLANCE PROGRAM MEASUREMENT DUKOVANY NPP

- 2 O-wires (container rotation against core, determination of azimutal fluence distribution in container)
- 3 I-wires (determination of axial fluence high distribution in container)
- 2 spectrometric sets (determination of neutron fluence spectrum)
- Programs BASA CF, SAND and relative fluence distribution model is used
- Fluence is determined in the container axis in axial container centre and in geometric centre of individual specimens
- Input spectrum from 1-M-1 and 1-M-2 measurement in surveillance program

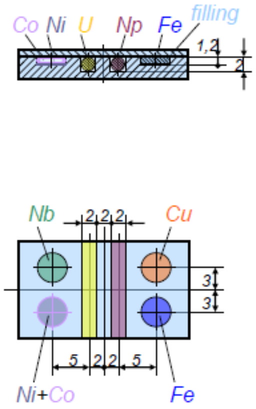




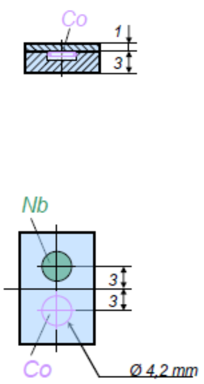
Interní

FLUENCE – SURVEILLANCE PROGRAM MEASUREMENT TEMELIN NPP

- **Cu, Fe wires in container cap**
- **12 spectrometric sets in stainless steel capsules per container**
  - 8 large spectrometric sets
  - 4 small spectrometric sets
- **Large spectrometric set**
  - Consists of Co, Cu, Fe, Nb, Ni detectors; metal discs ∅4 x 0,1mm
  - Large sets in bottom part of container contain fission detectors NpO_2, UO_2
- **Small spectrometric set**
  - Consists of Co, Nb detectors ; metal discs ∅4 x 0,1mm



Large spectrometric set

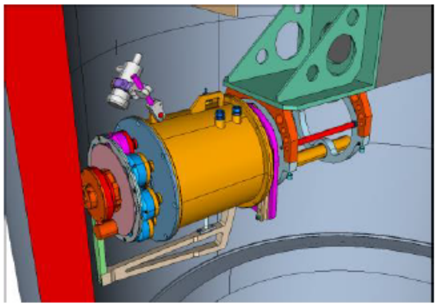


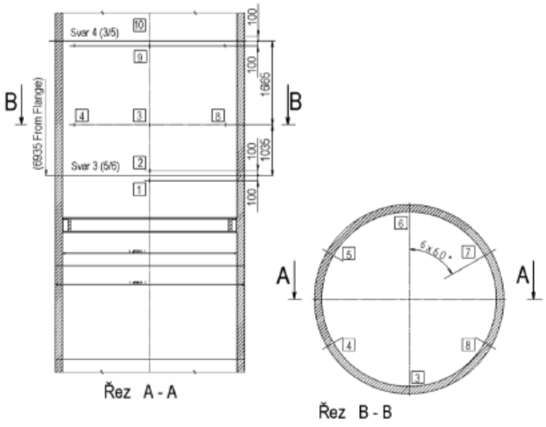
Small spectrometric set

Interní

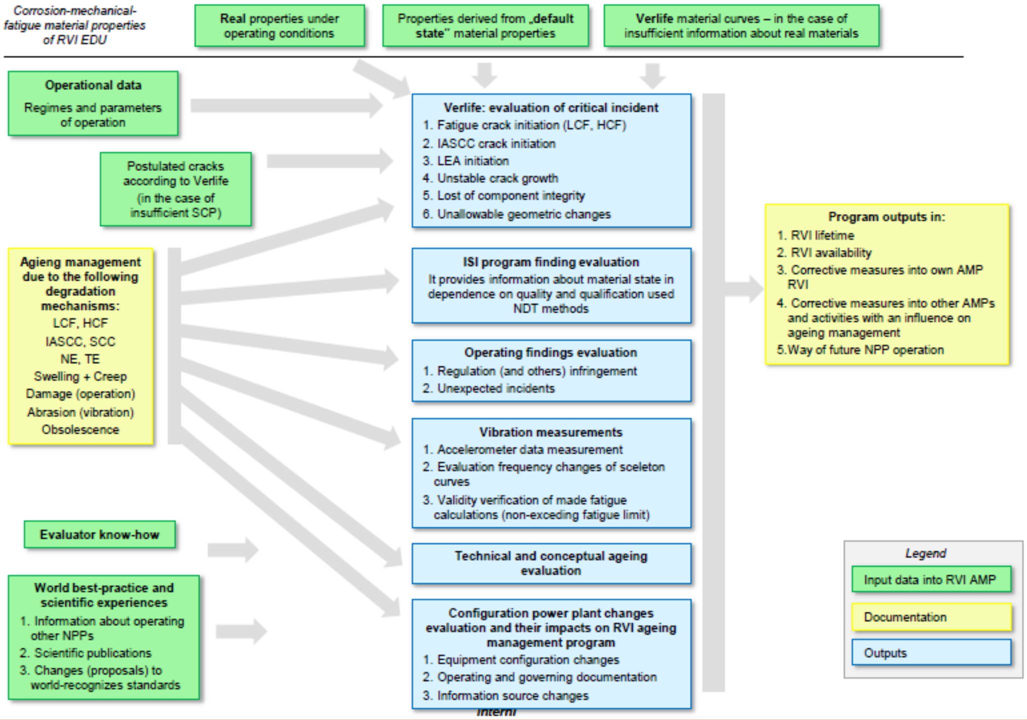
SAMPLES FROM RPV CLADDING

- Samples of Dukovany RPV cladding were taken from 3th block in 2005 and 2017
- It was taken 8+16 samples in total
- Goal: qualification of fluence evaluation





SPECIFIC AMP FOR RADIATION DAMAGE OF ITNERNALS – UNDER CONSTRUCTION



Corrosion-mechanical-fatigue material properties of RVI EDU

Real properties under operating conditions

Properties derived from „**default state**" material properties

Verlife material curves – in the case of insufficient information about real materials

Operational data

Regimes and parameters of operation

Postulated cracks according to Verlife (in the case of insufficient SCP)

Agieng management due to the following degradation mechanisms:
LCF, HCF
IASCC, SCC
NE, TE
Swelling + Creep
Damage (operation)
Abrasion (vibration)
Obsolescence

Evaluator know-how

World best-practice and scientific experiences
1. Information about operating other NPPs
2. Scientific publications
3. Changes (proposals) to world-recognizes standards

Verlife: evaluation of critical incident
1. Fatigue crack initiation (LCF, HCF)
2. IASCC crack initiation
3. LEA initiation
4. Unstable crack growth
5. Lost of component integrity
6. Unallowable geometric changes

ISI program finding evaluation

It provides information about material state in dependence on quality and qualification used NDT methods

Operating findings evaluation
1. Regulation (and others) infringement
2. Unexpected incidents

Vibration measurements
1. Accelerometer data measurement
2. Evaluation frequency changes of sceleton curves
3. Validity verification of made fatigue calculations (non-exceding fatigue limit)

Technical and conceptual ageing evaluation

Configuration power plant changes evaluation and their impacts on RVI ageing management program
1. Equipment configuration changes
2. Operating and governing documentation
3. Information source changes

Program outputs in:
1. RVI lifetime
2. RVI availability
3. Corrective measures into own AMP RVI
4. Corrective measures into other AMPs and activities with an influence on ageing management
5. Way of future NPP operation

Legend

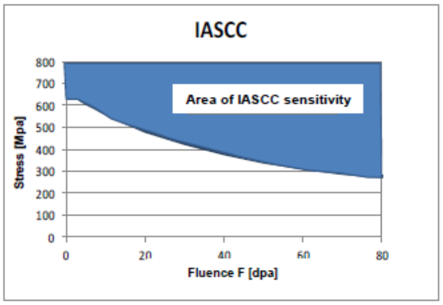
🟩 Input data into RVI AMP

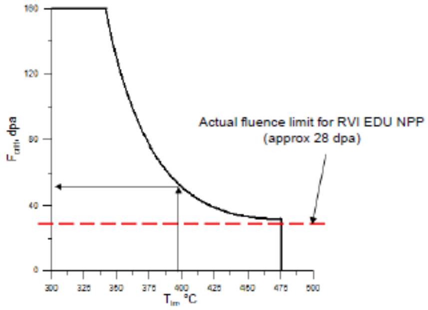
🟨 Documentation

🟦 Outputs

LIFETIME ASSESSMENT OF RVI

- **Lifetime Assessment of RVI Report**
  - ➢ lifetime assessment according to IAEA Verlife Appendix C – degradation mechanisms – low cycle fatigue, high cycle fatigue, IASCC, LEA, swelling, creep





C Critical curve for LEA estimation according to VERLIFE.

THANK YOU FOR YOUR ATTENTION

ANY QUESTIONS ?

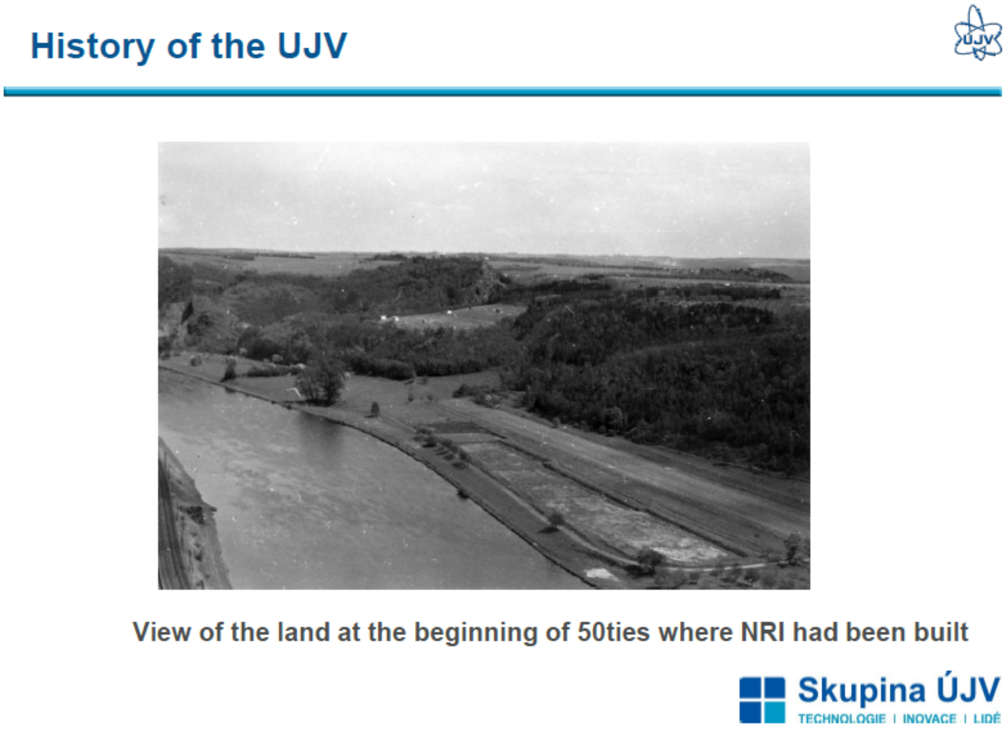
B.8 UJV Activities in International Research Projects in the Field of RPV and RVI (M. Zamboch)

ÚJV Řež, a. s.

UJV Activities in International Research Projects in the Field of RPV and RVI

Nuclear Regulatory Commission
International Workshop on Age-Related Degradation of Reactor Vessels and Internals
May 2019

Miroslav Žamboch,



History of the UJV

View of the land at the beginning of 50ties where NRI had been built

History of the UJV

Building of the UJV in late fifties(founded in 1955)

Skupina ÚJV
TECHNOLOGIE | INOVACE | LIDÉ

UJV from opposite bank of the Vltava river

Skupina ÚJV
TECHNOLOGIE | INOVACE | LIDÉ

UJV activities

Services for Czech NPP EDU and ETE

Preparation of building of new nuclear power plant in Czech republic

Renewal of fossil power plants

Nuclear medicine (diagnostic, cancer medicals)

SOTERIA

Safe long term operation of light water reactors based on improved understanding of radiation effects in nuclear structural materials

- **Funded by European Commission under Horizon 2020 research and innovation programme**
- **24 participants**

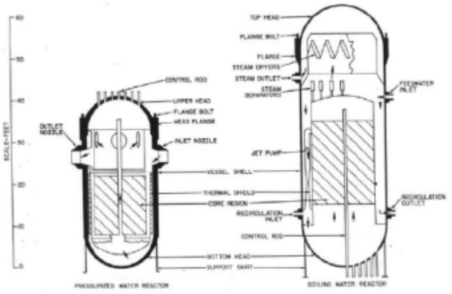


FIG. 6. Comparison of PWR and BWR RPVs with the same output.

TYPICAL PWR RPV

- Vessels may weigh up to 800t with wall thickness up to ~330mm.

WWER-440 RPV

SOTERIA

Scope of research

- **Experimental work focused on flux and fluence effect on RPV and RVI**
- **Effect of microstructure and composition on residual lifetime of the RPV,**
  - Development of synergic fracture toughness prediction formulas
- **Effects of chemical and radiation environment on RVI**
  - Base for developing of predictive tools for end users
- **Development of models for the assessment of ageing mechanisms in RPV and internals, setup of the platform for modelling**

- **Education of the nuclear engineer and nuclear community**

Age60+

The goal is to improve the access of workers across Europe to useful data on NPP components ageing which might be unpublished or published obscurely

- **Partly funded by the European Atomic Energy Community's (Euroatom) Seventh Framework Programme**
- **Encourage European researchers to share data in order to maximize its utility**
- **Consolidate the data in accessible formats**
- **Utilise selected accessible data to assess the applicability of current methodologies to cover 60+ year of operation**
  - To produce a new embrittlement trend curve for VVER 440 RPVs
  - To identify factors influencing embrittlement that are not properly described in current ETC for MnMoNi RPV steels
  - Identify method to reducing measurement uncertainty in the radiation induced shift of Charpy ductile to brittle transition temperature
  - Suggested improvements to Charpy measurement protocols
  - Provided a proof of principle for a more robust derivation of ETCs from Charpy or fracture toughness measurements

INCEFA, INCEFA+

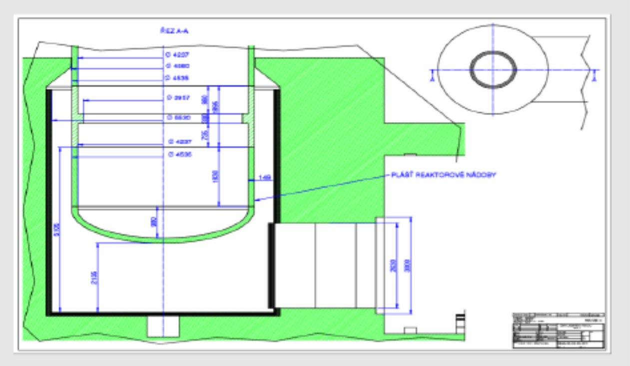
Increasing safety in NPPs by Covering gaps in Environmental Fatigue

- **Partly funded by the Euratom Research & Training programme**
- **Under NUGENIA Association**
- **16 participants**
- **Summarized state of art of environment influence on fatigue life evaluation**
- **Fatigue experimental program to refine the evaluation and prediction models and formulas**
  - Agreed testing protocol
  - LWR environment & air
  - SS304 common material, national materials
- **Development of the Fatigue Assessment Procedure**
- **Dissemination and Training**

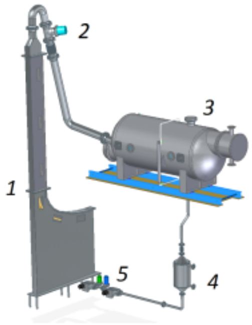
THS 15 – In Vessel Retention Research

Justification of IVMR strategy for VVER reactors 1000/320

- **100 small scale experiments**
- **Requirement (necessity) for large scale experiments with fully justified geometry**
- **Developed and assembled in UJV Řež, a.s.**
- **Tests of surface influence on cooling**



VVER 1000 cavity configuration



3D model of the cooling channel and primary circuit

THS 15 – In Vessel Retention Research, cont.

From instalation of the equipment …

Installation – because of the dimension through the roof

View of the condenser and upper part of the cooling channel

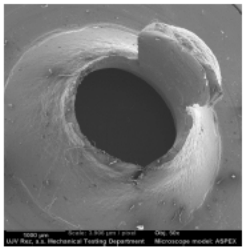
Defi - Prosafe

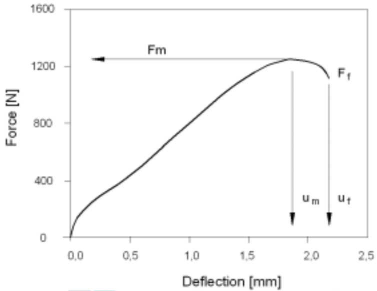
Benchmark focused on probabilistic assessment of RPV integrity

- **Partly funded by the European Atomic Energy Community's (Euratom) Seventh Framework Programme FP7/2007-2013 under grant agreement No. 604965**
- **In Europe the resistance against fast fracture based upon deterministic approach, probabilistic voluntary (supplementary)**
- **Several benchmarks project performed in the past with unsatisfying results and questions still opened, Defi – Prosafe project is continuation of former work performed**
- **The objectives are to support utilities in the assessment of the regulatory margin justification in their structural integrity assessment of the RPV by demonstrating a low risk of sudden failure in the case of a request for long-term operation as well as to progress on the acceptance within Europe to use a probabilistic approach for integrity.**
  - Determination of the limiting material reference temperature based upon RPV defects distribution, material properties distribution, and TH uncertainties

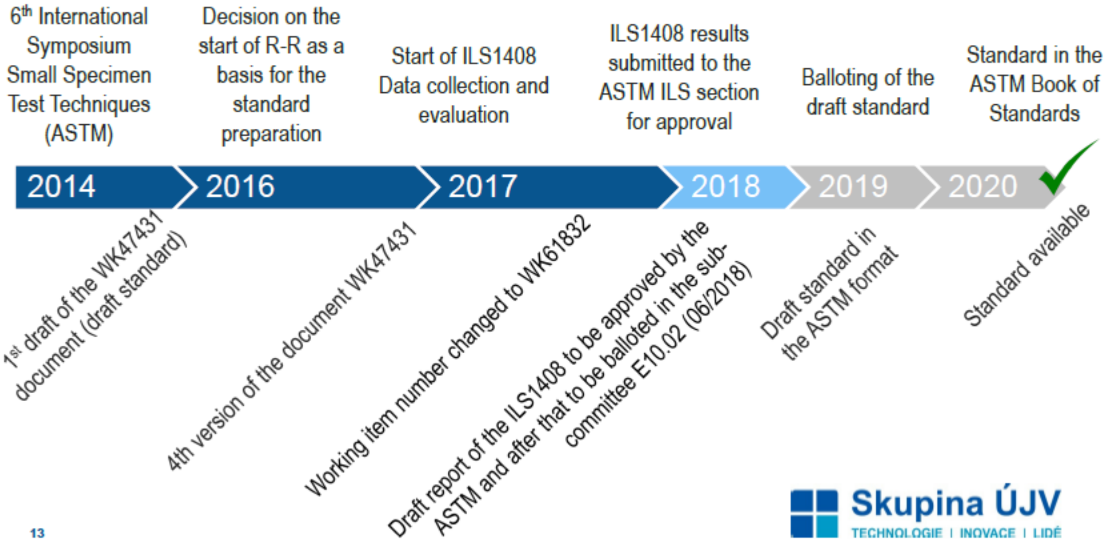
Project of Small Punch Test Standardisation within ASTM

- **There does not exist standardised procedure for very small specimen that can by used for evaluation of mechanical properties specially in nuclear industry**
- **Within American Society for Testing and Materials the procedure developing project was started:**
- **Based upon interlaboratory study with selected material ILS1408**
  - 14 laboratories from Europe, Asia, USA
- **Continuation with tests of 6 selected materials – 7 states**
- **8 parameters recorded during the tests**





- **Expected schedule (after E10 committee meeting in June 2018)**



6th International Symposium Small Specimen Test Techniques (ASTM)

Decision on the start of R-R as a basis for the standard preparation

Start of ILS1408 Data collection and evaluation

ILS1408 results submitted to the ASTM ILS section for approval

Balloting of the draft standard

Standard in the ASTM Book of Standards

2014 2016 2017 2018 2019 2020

1st draft of the WK47431 document (draft standard)

4th version of the document WK47431

Working item number changed to WK61832

Draft report of the ILS1408 to be approved by the ASTM and after that to be balloted in the sub-committee E10.02 (06/2018)

Draft standard in the ASTM format

Standard available

NURESIM, NURISP, NURESAFE

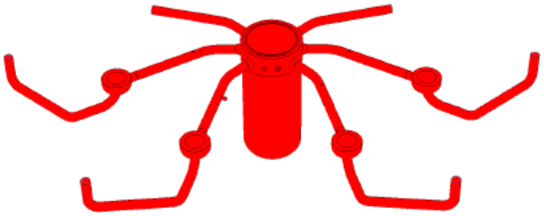
European platform for nuclear reactor simulations

- **Generally describing mixing phenomena relevant for safety analysis and particularly for structural integrity evaluation**
- **Development of the computer fluid dynamic simulation NEPTUNE**
- **The data from commissioning test used (Sizewell-B for PWRs, Loviisa and Paks for VVER), TOPFLOE, ROSA**
- **Recommendation for applicability of CFD codes for turbulent mixing problem**
- **Improving and validation simulation tools for modelling scenarios relevant to safety analysis of LWR (LOCA, PTS …)**
- **Multi-scale analysis and multiscale coupling of thermal hydraulics tools with others disciplines to investigate safety issues Pressure Thermal Shock (PTS), Critical Heat Flux (CHF), Loss of Coolant Accident (LOCA)**

Safety Analysis of VVER NPPs

Continual development, verification of new techniques / tools in safety analysis of the PWR/VVER reactors ,

- **As rupture of the RPV due to PTS and consequent LOCA is outside design accidents**
- **Continual process since 1999 (first analysis)**
- **Wider deployment of CFD in PTS analysis**
- **Verification of CFD results with respect to the older simple mixing results**
- **As results the less overconservative modelling of TH processes are acquired as important input for following RPV/RVI evaluation**
- **Still open question – CFD modelling of two phase cases scenarios**

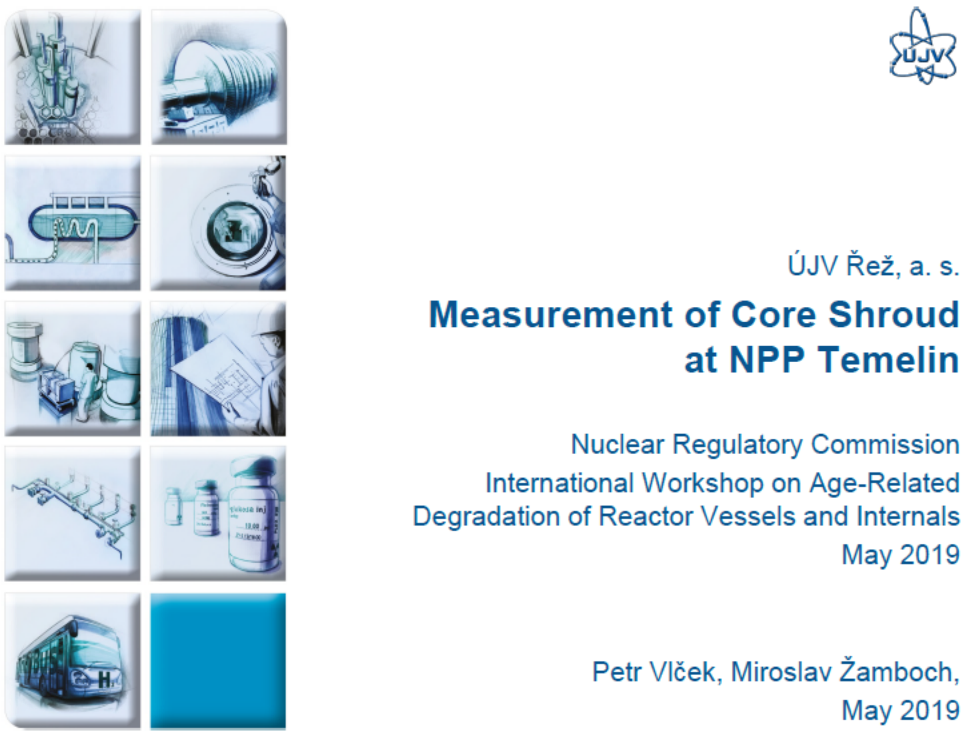


Computational Domain – CFD model VVER 440

- 2.1M computational cells, 1.4M cells in fluid domain, 0.7M cells in solid walls
- Calculations of long transients (~1hour or longer)
- Initial and boundary conditions for CFD are taken over from RELAP5 simulation.
- Goal of the CFD simulation: temperature fields on wetted walls in cold legs and on RPV wall in downcomer
- Depending on the solved case, some parts can be deleted from the computational domain, e.g. cold legs without operating injections.

Thank you!

B.9 Measurement of Core Shroud at NPP Temelin (M. Zamboch)

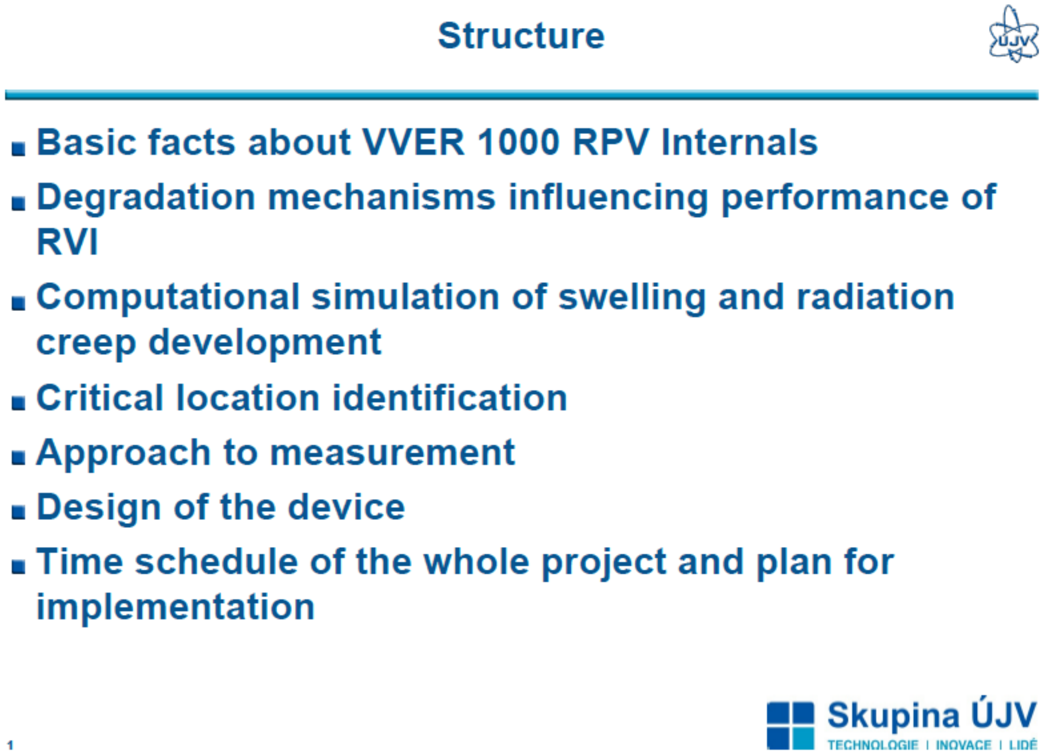


ÚJV Řež, a. s.

Measurement of Core Shroud at NPP Temelin

Nuclear Regulatory Commission
International Workshop on Age-Related
Degradation of Reactor Vessels and Internals
May 2019

Petr Vlček, Miroslav Žamboch,
May 2019



Structure

- Basic facts about VVER 1000 RPV Internals
- Degradation mechanisms influencing performance of RVI
- Computational simulation of swelling and radiation creep development
- Critical location identification
- Approach to measurement
- Design of the device
- Time schedule of the whole project and plan for implementation

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NPP Temelin Basic Information

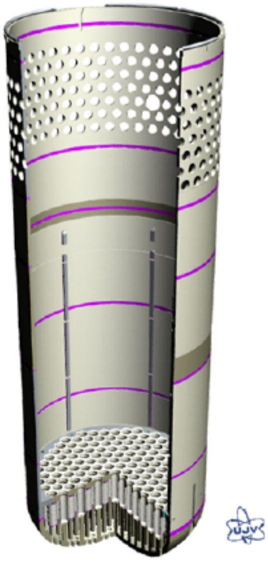
VVER 1000 type, (PWR) 2 units

Russian design, manufactured in Czech republic by Czech companies

- Operation since 2001
- Power up-rate to 105.7 % - 1056 MW of electric power output
- Original design life time 30 years (excluding RPV which has 40 years)

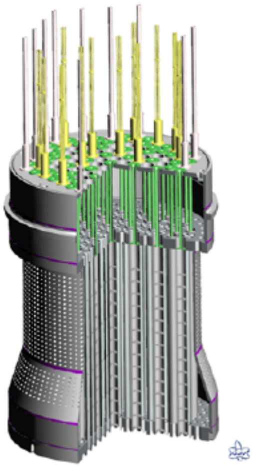
RPI VVER 1000 – Components and Material

- **Basic material is Ti stabilised austenitic stainless steel 08Ch18N10T (similar to A321]**
- **Small parts are from hi strength Fe-Ni-Cr alloy (ChN35VT-VD)**



Core barrel

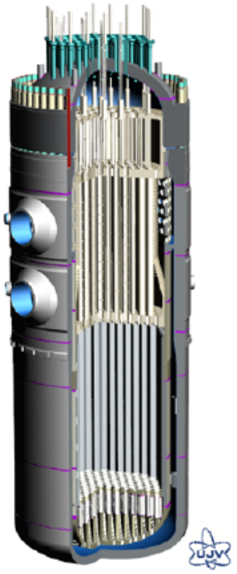
Core shroud



Protective tube unit

Functions of RPV Internals

- Support and geometry maintenance of the core
- Medium flow control in RPV
- Control elements functions support
- Instrumentation equipment support
- Shielding of the RPV material from neutron and gamma radiation



Chemical composition of RVI main construction materials

Ocel	C	Mn	Si	P	S	Cr	Ni	Ti	W	Co
08Ch18N10T	0.07	1.42	0.44	0.020	0.010	18.65	10.50	0.50		
ChN35VT-VD	0.08	1.48	0.43	0.015	0.012	14.88	34.88	1.39	3,38	0.02
14Ch17N2	0.16	0.40	0.40	0.022	0.007	16.32	1.95			0.02

Design and Operational Parameters

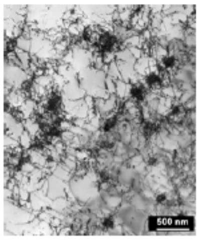
- Design pressure up to 17.65 MPa
- Design Temperature up to 350°C, working temperature 320°C
- VVER chemistry (low oxygen, low hydrogen water with injection of boric acid)
- Radiation load up to 1.5 dpa/ year for most critical part of core shroud (computational prediction), it corresponds about 90 – 100 dpa after 60 years of operation

Expected and Controlled Degradation Mechanisms

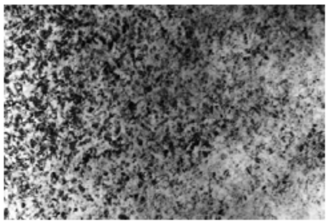
- **Fatigue (low and high cycle)**
- **IASCC**
- **Wear**
- **Radiation swelling and radiation creep**
- **Loss of fracture toughness due to thermal and neutron aging - Limit Embrittlement Area development**
- **Mechanical damage**

Swelling and radiation creep damage manifestation

- **Change in microstructure (Frankel loops, dislocation density increase, nanocavities, microcavities)**
- **Degradation of mechanical properties**
  - Elongation, Fracture toughness, Yield strength, Tensile strength
- **Increase in volume of heavily loaded area of material (neutron flux & temperature)**
  - As consequence – macroscopic (measurable by NDT method) change in geometry of RVI components: Core shroud



Dislocation structure of 08Ch18N10T steel prior irradiation



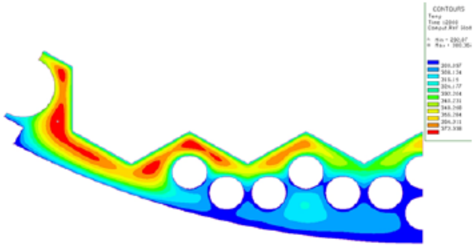
Irradiation defects in 08Ch18N10T steel after irradiation, Frank loops, perfect dislocation loops,"black dots".

Radiation Swelling and Radiation Creep of Core Shroud

- **Degradation mechanism "radiation swelling„ was detected in core baffle of type of VVER 1000**
  - Ukraine
  - Russia

- Degradation mechanism caused by a combination of fluence (dpa) and temperature (elevated temperature during operation plus heat from gamma radiation) distributed in the bulk of the core shroud.
- Degradation of the crystal lattice , misplaced atoms, micro-void, voids, change of fracture mechanical properties

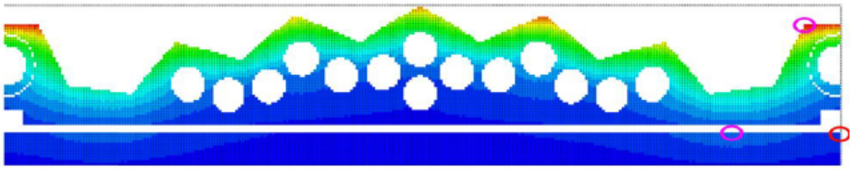
Calculated temperature distribution for the 12th campaign of the 1st block of NPP Temelin.



- **In CR today's assessment of swelling and radiation creep - only computational**

Identification of Critical Point of Swelling Development

- **First calculation of RS and RC development were performed in 2012 – 2013 by NRI Rez and Gidropress (Russia) independently with good agreement**
- **Actually the absorbed dose and temperature distribution is calculated for each campaign to cover complex history of temperature & neutron load as key factor of swelling development**

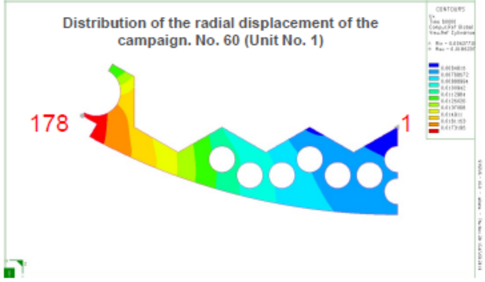


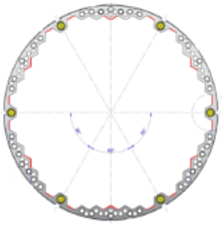
2D calculation model (core shroud & barrel) with identification of location with maximal dpa and gamma heating values in selected campaign, 2018 calculation

Identification of Critical Point of Swelling Development

Based on the calculation the most critical locations of core shroud swelling (and consequently geometry changes) are:

- The center of the segment (point no. 1) moves according to the simulation towards the core
  - The gap between inner surface of core shroud and fuel element is 4 mm.
- The outer surface of the core shroud where threaded rod is located point no. 178, it moves according to simulation in direction to the barrel
  - The smallest gap between core shroud and barrel is 2.5 mm (up to 10 mm).


Distribution of the radial displacement of the campaign. No. 60 (Unit No. 1)



Measurement of Core Baffle

Benefits of the core baffle assessment using NDT - dimensional inspection

- Provides the information about the actual state of component and demonstrate the functionality of the device for the next operation
- Repeated measurement give information about dimension changes, and makes possible experimental trending and prediction of the DM „radiation swelling“ development
- Verify calculations of DM „radiation swelling“ development, calibration of the physical model used
- Make possible to plan corrective measures well in advance of their actual implementation,
- Verification of corrective actions.

Experience with measurement of Core Shroud in the world

- **Russia**
  - Computational evaluation of swelling is compulsory, experimental verification recommended for decrease of conservatism
  - Macroscopic changes of core shroud geometry measured
  - Laser scanning approach, the scanning probe positioned in the *center reactor point* at the different levels of Z- axis
  - Declared level of accuracy 0,5 mm

- **Ukraine**
  - Experimental verification of the core shroud geometry changes are required for LTO phase for VVER 1000
  - Macroscopic changes of core shroud geometry measured
  - UT scanning approach used, the UT probe on the beam centered with respect to the centre reactor point at the different levels of Z- axis

---

UJV approach to measurement of Core Shroud dimension changes

The basic requirements:

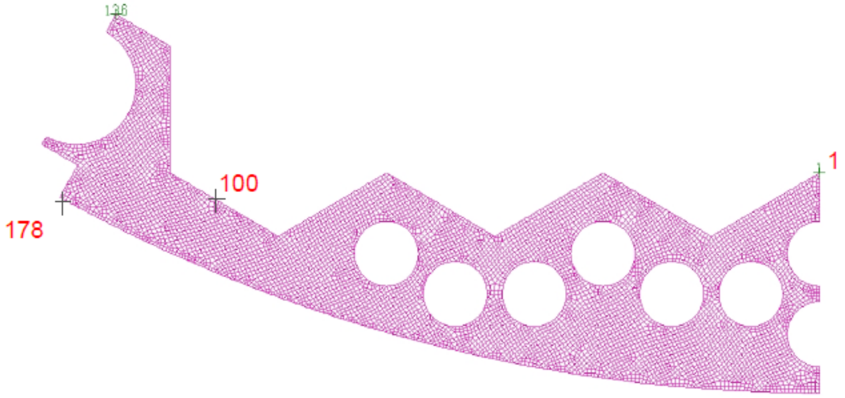
- **The measurement must not be dependent upon *center point of the reactor* as it was identified as the most problematic part of measurement realized**

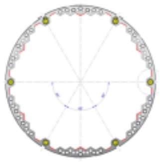
- **It should be possible to realize measurement with the same equipment during time horizon 15 years**

- **The device must be independently calibrated prior each measurement**

Selection of locations for measurement

The positions of selected nodes on the "inner surface" of core baffle with large displacement during operation due to swelling development





Node no. 1 - near of two small canals.

Node no. 100 - in the middle of the facet near the grand canal.

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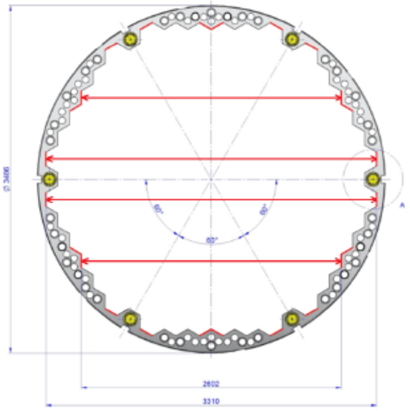
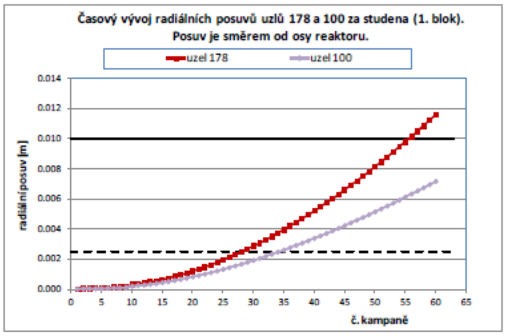
UJV approach to measurement of Core Shroud dimension changes, cont.

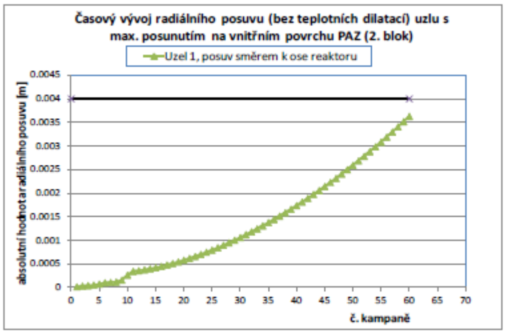
The measurement principle

- Distance measurement of specific locations at the inner surface of the core shroud as it is specified in the picture (areas near threaded rod and the middle segment).
- Device rotation 2x 60 ° should provide a information about geometry changes of the whole perimeter of the core shroud ring.
- The measurement is performed at the different Z axis level (at least 3 per core shroud ring, or as required).
- Advantage – the specified distance were measured during assembly process of the reactor internals so zero values are accessible for comparison



The predicted shifts of selected nodes





Prediction of displacement for nodes No. 100 and 178. It is necessary to highlight that prediction is based upon the worst campaign from neutron flux point of view. Displacement prediction based upon real history shall be more modest.

Displacement of Node. 1 is about 3,6 mm after 60 years, about 0,6 mm after 20 years.

The measured change of diameter after 20 years should be 1,2 mm (in the direction to the axis of the reactor).

Hardware requirements

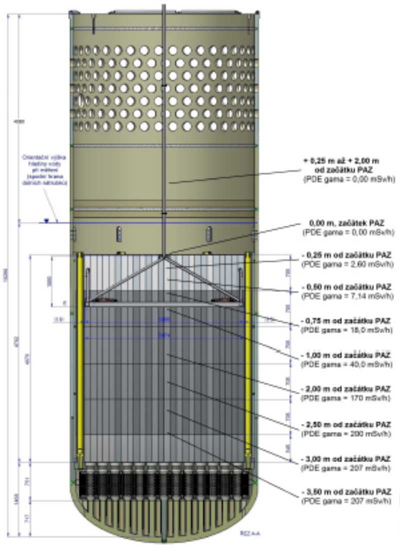
Requirements for inspection implementation

- **Measurements in the water with boride acid**
- **Under hydrostatic pressure corresponding to the depth of reactor shaft.**
- **High level of radiation.**
- **Coolant temperature - maximum 50 ° C.**
- **It is necessary to minimize influence on length of NPP outage.**
- **Requirements for a high measurement accuracy – order of tenths of mm.**

Gamma radiation dose measurement

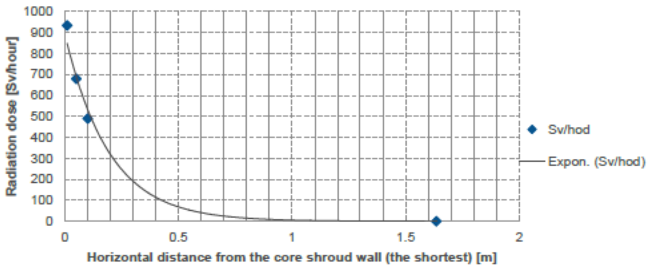
To guarantee life time of the device it was necessary to determine the radiation dose and qualify resistance of the material and component used

Measured gamma values (mSv/h) In the centre of CS	Z axis level with respect to the CS “beginning” [m]
0	-2
0	-1
0	-0,5
0	-0,25
0	0
2,6	0,25
7,14	0,5
18	0,75
40	1
170	2
200	2,5
207	3
207	3,5



Gamma radiation dose measurement, cont.

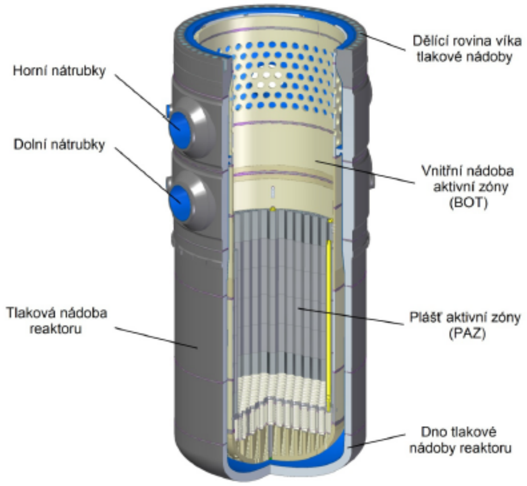
Computational simulation of radiation dose as function of the distance from the core shroud wall in the middle z plane

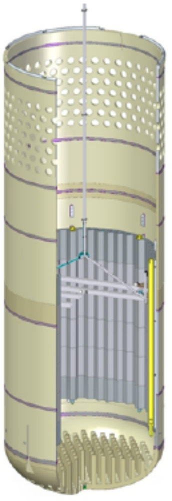


As water provide very good shielding it was decided that where possible, the components sensitive do radiation shall be positioned in the centre of the CS

Measurement of Core Shroud – Design Model

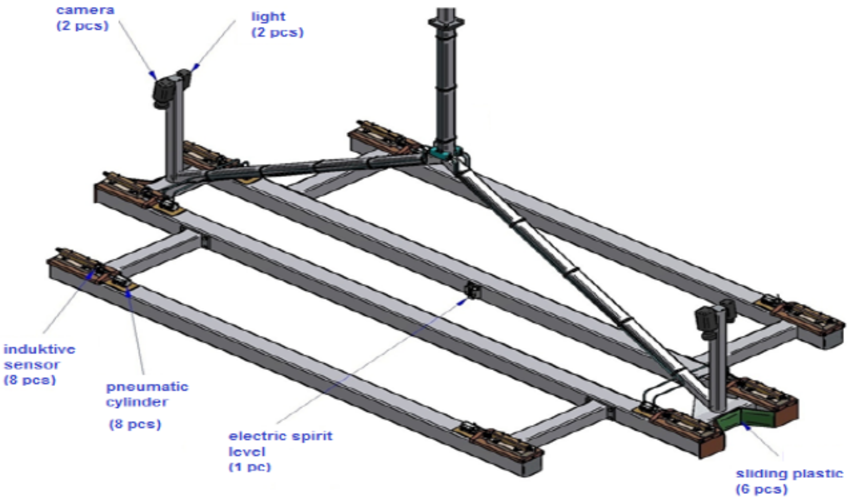
Reactor vessel, model design of the device in core shroud





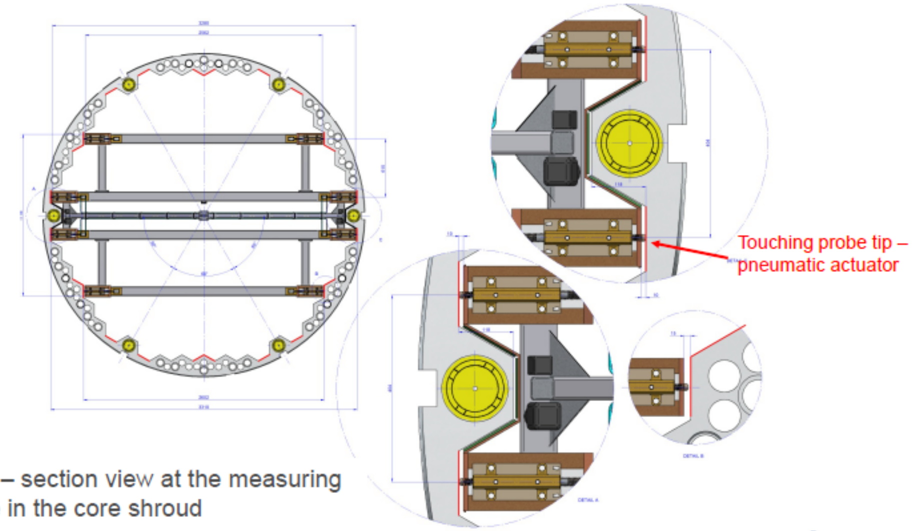
Measurement of Core Baffle – Design Model

Components of the device:



Measurement of Core Baffle – Design Model

Drawing of the device for the core shroud baffle dimensions measurement.


Touching probe tip – pneumatic actuator

Cross – section view at the measuring device in the core shroud

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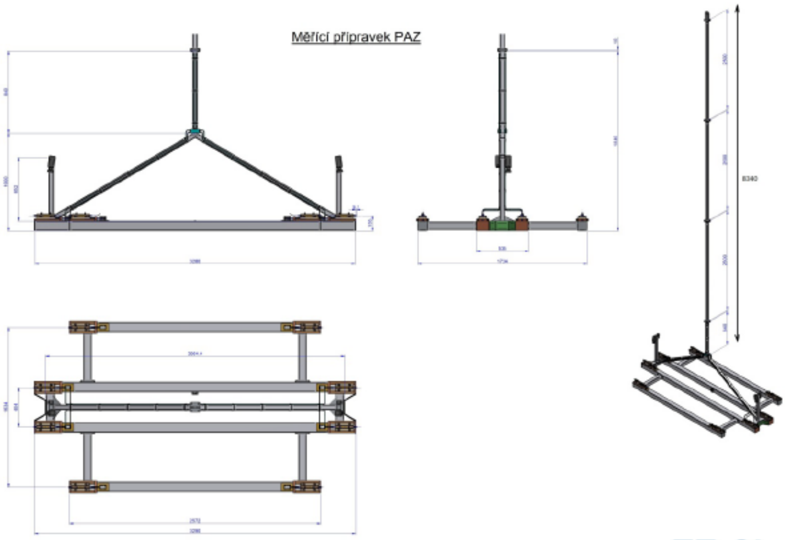
Measurement of Core Baffle – Model of Design

Drawing of equipment for measuring



Design - accuracy

Description of the sensors used (including parameters)

- Incremental induction sensors (touching the surface of the core shroud):
  Displacement in the range of 0-15 mm

- Optical length sensor – optical fibre:
  Measured dimension: 2 x 2602 mm and 3310 mm x 2

- Electronic water level (spirit level) (ACS-360-1-SC00-VK2-PM)
  Measuring range: 360 °
  Tilt measurement accuracy: ± 0.1 °

- Thermometer for compensation of temperature changes:
  Measurement range: 20 to 40 ° C (max. 50 ° C)

- The entire system:
  Officially declared accuracy during the entire lifetime: ± 0.25 mm (supposed ± 0.1 mm)

Design Model - Requirements

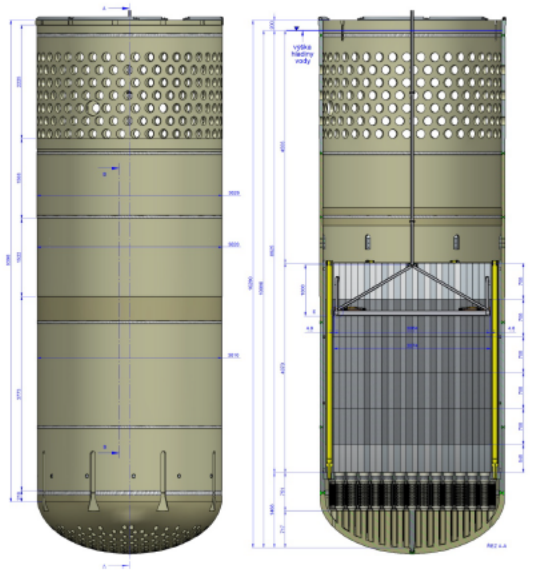
Specific hardware requirements

- Measurement frequency every 3 years
- Lifetime of the equipment at least 20 years.
- Environment about 10 m below the water surface with the admixture of boric acid (H3BO3) in a concentration of 15 g / l.
- Equipment must withstand ionizing radiation.
- Incremental optical sensors must allow pneumatic ejection test probes (to ensure an adequate pressure during measurement) and subsequent retraction of the probe spikes.
- In case of accidental failure of the system must ensure the automatic retraction of probe tips using (springs) to avoid jam of the inside the core shroud.

The storage and transport dimensions are limited by requirement of NPP to (L x W x H): 3300 x 600 x 1100 mm

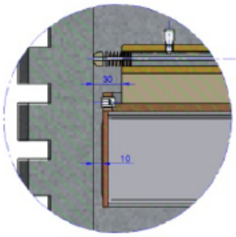
Measurement of Core Shroud – Design Model

- Position of device in core barrel during measurement

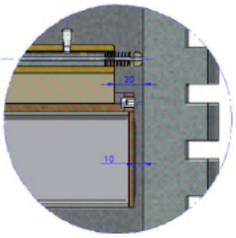


Measurement of Core Shroud

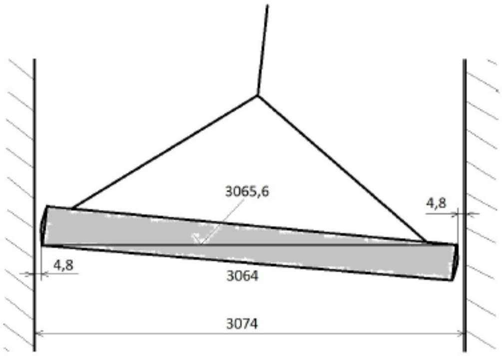
- Analysis of the possibility of a jam of the measuring equipment core baffle

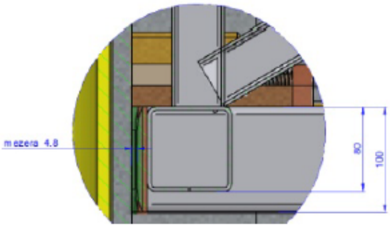


DETAIL C



DETAIL D





Due to rounding of the guiding sides of the measurement platform the smallest possible clearance during tilting accident is 9.6 mm.

Measurement of Core Baffle

Conditions for the implementation of controls at ETE

- **The best possibility - during overhaul with repair of MCP**
  - Minimal influence on the critical path - about 3-4 days during ISI of RPV flange, so called period of "axis of cold nozzles„.
  - The water level in the reactor low "on the axis of cold nozzles„.
  - Water temperature about 35-40 °C.
  - High radiation.

Polar crane in reactor hall as tool for measurement of core shroud dimension

Measurement of Core Shroud

Time shedule of measurement

- **The equipment is going to be transported in advance to the reactor hall for preparatory work before the test.**
- **A rough estimate of the time required for the proposed measurement procedure is:**
- mounting on the crane, transport of equipment to core baffle = 1.5 hours
- the measurement = 6 - 10 hours
- removing the equipment, releasing the crane = 1 hour
- a total of up to about 12 hours
- **After detachment from the crane the device must undergo complete decontamination and packaging for storage.**
- **According capabilities and available capacity crane equipment is going to be loaded onto a carriage and transported out of the reactor hall.**

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Measurement of Core Shroud

The project schedule

2016 - Engineering design of measuring equipment

- structural design measuring instrument
- development of measurement technology and data processing

2017 - Design and manufacture of various parts of the equipment, qualification of measuring equipment

2018 - Manufacturing of the prototype, production test specimen for qualification, initiation tests to verify the measurement technology

2019 - Qualification and calibration of measuring equipment, completion of documentation

2020 - Submission of final documentation of equipment and documentation for measurement in NPP

2021 (2022 – 2023 according to NPP operation) - the realization of measurement

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Measurement of Core Shroud – Summary

Measuring is going to enable:

- To obtain information about the true state of the core shroud - detect deformation due DM „radiation swelling“.
- Repeated measurement to determine a trend of effect of DM radiation swelling on the geometry of core shroud.
- To verify calculations describing DM „radiation swelling“.
- To demonstrate the functionality of the device for further operation and to implement aging management program with respect to swelling.

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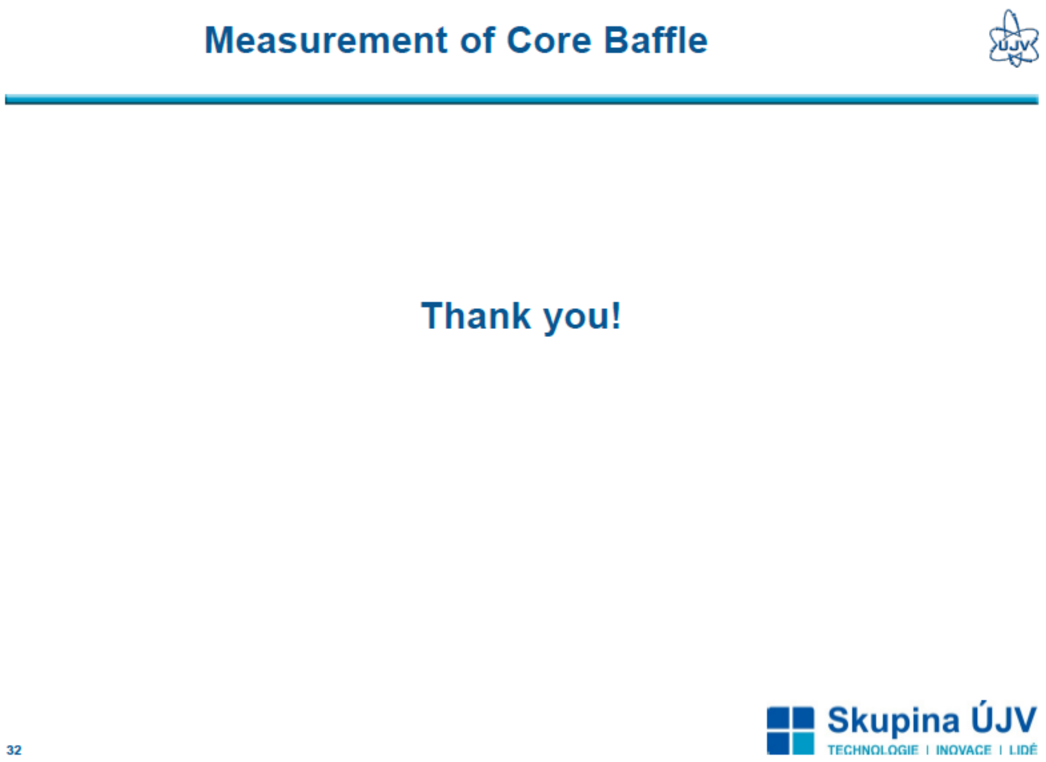

Measurement of Core Baffle

Thank you!

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B.10 Belgian R&D on Environmental Effects on Materials Degradation in LWRs (S. Gavrilov)

Belgian R&D on Environmental Effects on Materials Degradation in LWRs

International Workshop on Age-Related Degradation of Reactor Vessels and Internals, USNRC, Washington, 23-24/5/2019

SCK•CEN
STUDIECENTRUM VOOR KERNENERGIE
CENTRE D'ETUDE DE L'ENERGIE NUCLEAIRE

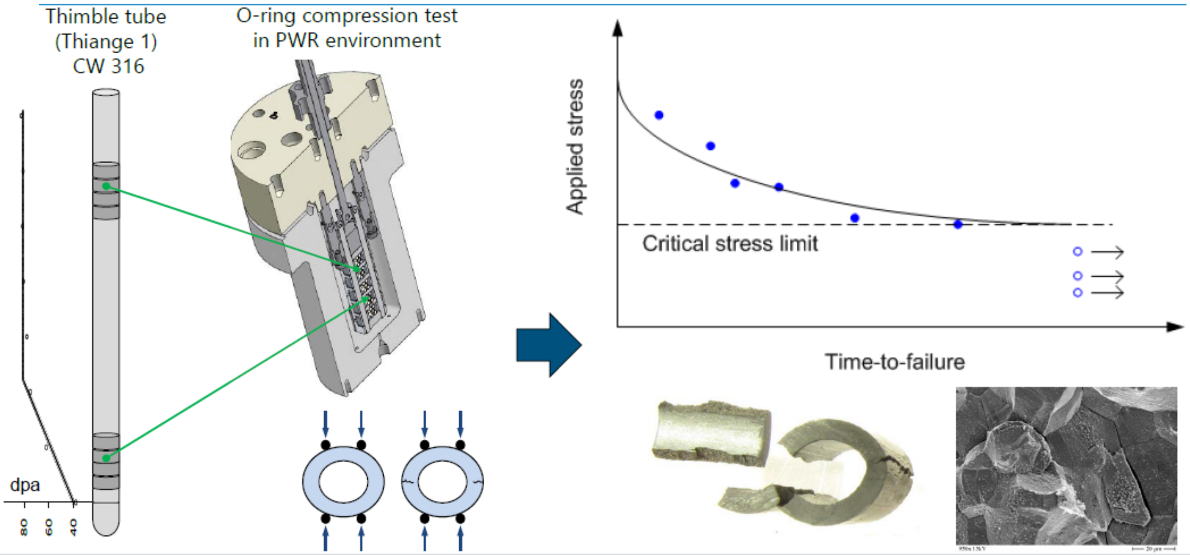
SCK•CEN/33980469

ISC: Public

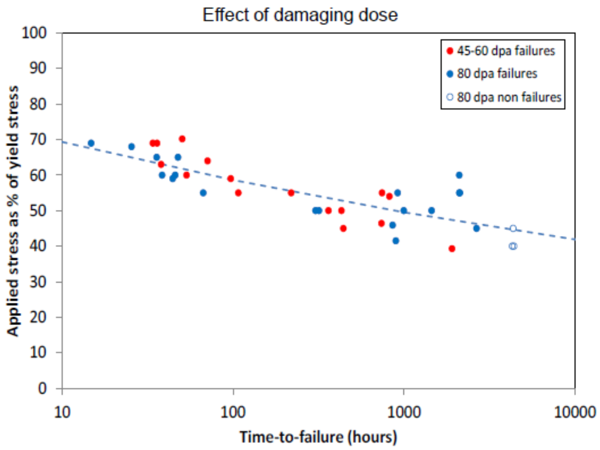
Outline

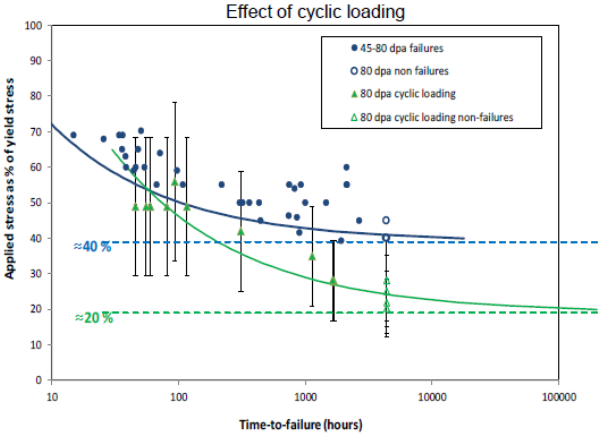
- IASCC
- Hydrogen issues for RPV
- Corrosion fatigue austenitic stainless steels in PWR environment

IASCC: experimental program



IASCC: Results





R. W. Bosch, M. Vankeerberghen, R. Gérard, F. Somville, IASCC crack initiation testing of thimble tube material with a dose up to 80 dpa under PWR conditions, Fontevraud 8, 2014

R. W. Bosch, M. Konstantinovic, M. Vankeerberghen, R. Gérard, F. Somville, Effect of cyclic loading on IASCC stress threshold of thimble tube material with a dose up to 80 dpa under PWR conditions, Fontevraud 9, 2018

IASCC: Outcome

- No apparent difference of time to failure – stress limit between 45 & 80 dpa -> saturation
- Stress threshold at ≈40 % of yield stress (≈400 Mpa) for time up to 6 months
- Cyclic loading failures occurred at a significantly lower average stress than under constant load conditions, effect was more pronounced at lower stress levels
- On-going: Development of deterministic & probabilistic failure assessments based on experimental results

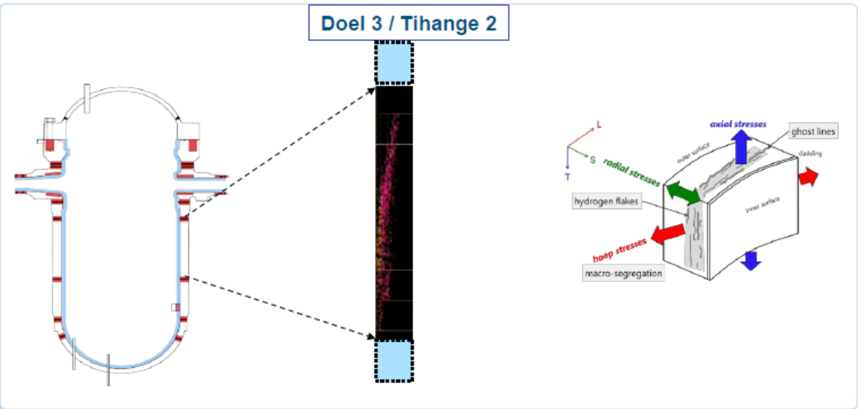
Hydrogen issues

2012 Discovery of flows in RPV of Doel 3 & Thihange 2 NPP identified later as hydrogen flakes formed during fabrication

2015 Hyphothesis of Prof. W. Bogaerts (KU Leuven) with support of Prof. D.D. Macdonald (UC Berkeley) on defects instability (media & press)

2016 Report of Boonen&Peirs on balance of H_2 in ingot: deficit of H_2 to form hydrogen flakes structure

2015-2018 Presentations, papers, invited talks, master thesis on “Mechanisms for Instability of Hydrogen Flakes During Reactor Operation”



2012 UT Doel 2/Tihange2

2014 UT Doel 2/Tihange2

2016 UT Doel 2

2017 UT Tihange 2

No evidences on further evolution of the defects during operation

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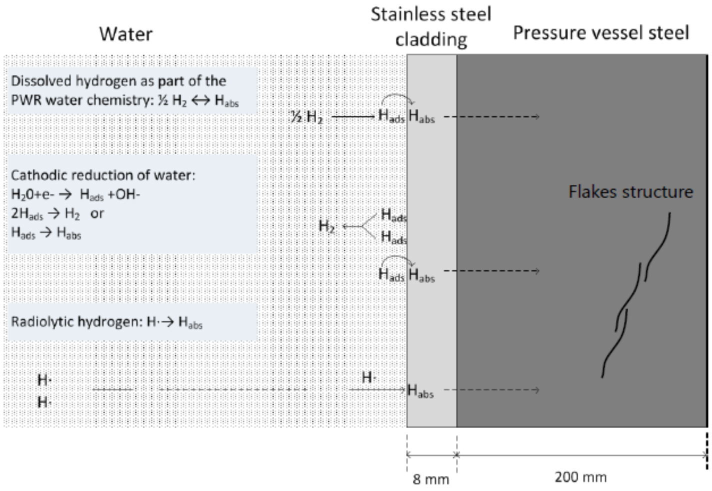
Hydrogen

Potential sources of hydrogen



Water

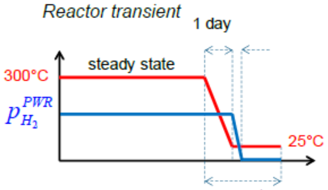
Stainless steel cladding

Pressure vessel steel

Dissolved hydrogen as part of the PWR water chemistry: ½ H_2 ↔ H_{abs}

Cathodic reduction of water:
$H_2O+e^- \rightarrow H_{ads} + OH^-$
$2H_{ads} \rightarrow H_2$ or
$H_{ads} \rightarrow H_{abs}$

Radiolytic hydrogen: $H\cdot \rightarrow H_{abs}$



Mass-transport model

- ☐ Atomic hydrogen

$$\frac{\partial c_H(x,t)}{\partial t} = -D_H\frac{\partial^2 c_H(x,t)}{\partial x^2} - S_H^{ft}(x,t) - S_H^{dt}(x,t) - S_H^{flake}(x,t)$$

- ☐ Flat traps

$$\frac{\partial c_H^{ft}(x,t)}{\partial t} = S_H^{ft}(x,t)$$

- ☐ Deep traps

$$\frac{\partial c_H^{dt}(x,t)}{\partial t} = S_H^{dt}(x,t)$$

- ☐ Flakes

$$\frac{\partial p_{H_2}^{flake}(x,t)}{\partial t} - \frac{p_{H_2}^{flake}(x,t)}{T(t)}\frac{dT(t)}{dt} = R\,T(t)\frac{\left(k_2^{gw}\left(c_H(x,t)\right)^2 - k_1^{gw}\,p_{H_2}^{flake}(x,t)\right)}{2}\left(\frac{S}{V}\right)_{flake}$$

- ☐ Additional source term

$$S_H^{flake}(x,t) = \left(k_2^{gw}\left(c_H(x,t)\right)^2 - k_1^{gw}\,p_{H_2}^{flake}(x,t)\right)\left(\frac{S}{V}\right)_{flake} f_v^{flake}$$

Significant scatter on mass transport, adsorption/desorption rates and hydrogen trapping parameters in steels reported in literature

Assessment of hydrogen sources, mass transport and pressure build-up

- Assessment of atomic hydrogen sources:
  - Absorption
  - Radiolysis
  - Corrosion reactions
- Mass transport model of hydrogen in RPV structure
- Assessment of pressure built-up during reactor transients
- Critical evaluation of hydrogen balance during flake formation

Consistent explanation of formation

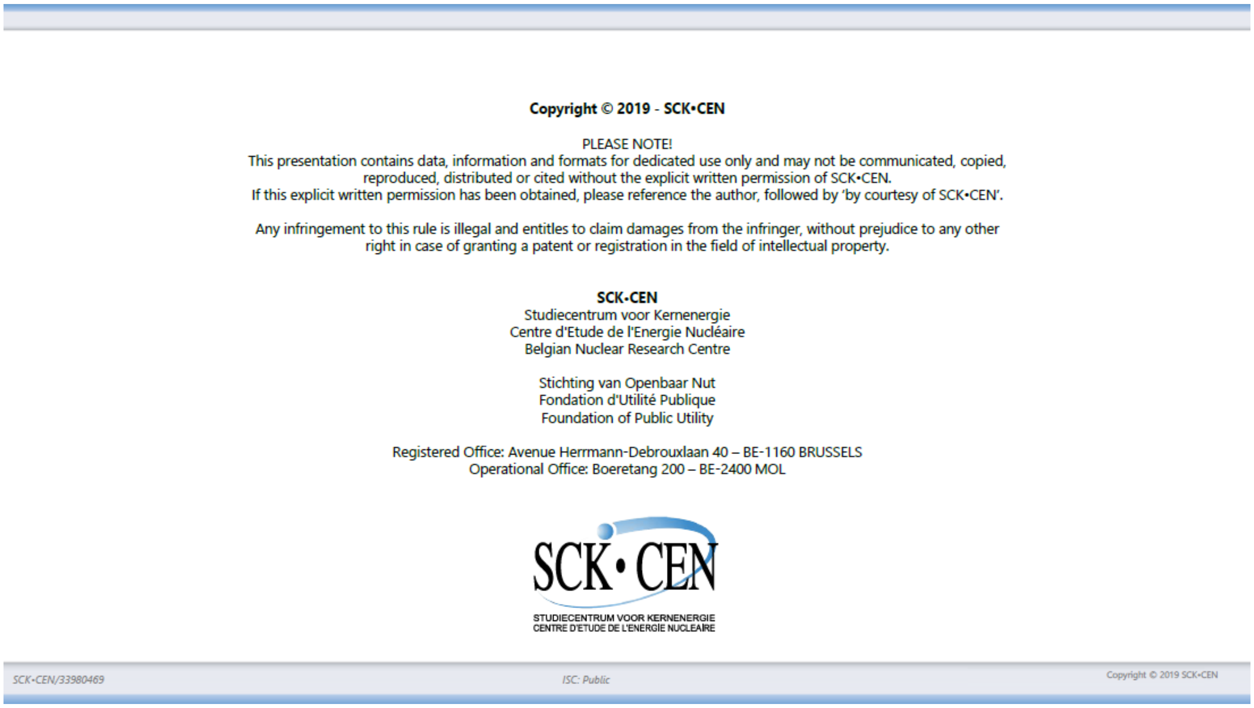
No driving force for flake propagation

- Experimental study in representative conditions: on hold

(Corrosion) fatigue

- Horizon 2020 INCEFA-PLUS (Increased safety in nuclear power plants covering deficiencies in the assessment of environmental fatigue) project 2015-2020
  - To address identified gaps and provide more data points on
    - Effects of surface conditions
    - Effect of hold time
    - Effects of mean stress/strain
  - Outcome
    - Protocol for Environment Assisted Fatigue testing (basis for ISO standard)
    - Modified fatigue analysis procedures (with more appropriate conservatism)
    - Fatigue curves for assessment of environmental effect on fatigue endurance (F_{en})
- Doel 1 Upper Plenum Injection line fracture examination (presentation "Doel 1&2 Upper Plenum Injection Line Issue", Michel DE SMET)

Vankeerberghen, M. et al., 2018. "Ensuring data quality for environmental fatigue – INCEFA-PLUS testing procedure and data evaluation", PVP2018-84081, Prague, Czech Republic
Bruchhausen, M., Mottershead, K., Hurley, C., Métais, T., Vankeerberghen, M., Cicero, R., Le Roux, J.C., 2017. "Establishing a multi-laboratory test plan for environmental fatigue", ASTM STP159820160047
M. Vankeerberghen, P. Marmy, L. Bens, PWR Fatigue Testing at SCK•CEN in the Framework of INCEFA+, Proceedings of the 7th International Conference on Fracture Fatigue and Wear, FFW 2018, Ghent, Springer



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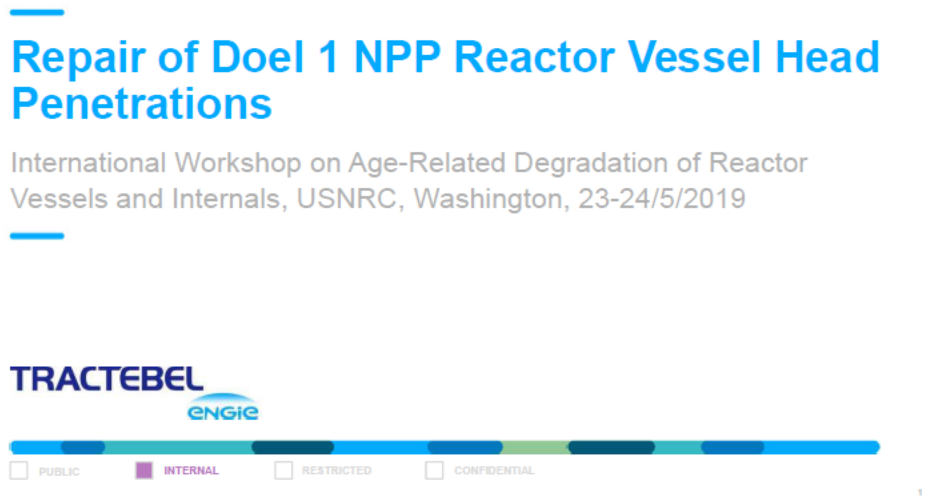
B.11 Repair of Doel 1 NPP Reactor Vessel Head Penetrations (C. Dupuit)



Repair of Doel 1 NPP Reactor Vessel Head Penetrations

International Workshop on Age-Related Degradation of Reactor Vessels and Internals, USNRC, Washington, 23-24/5/2019

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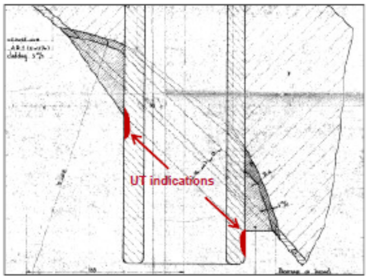
Repair Reactor Vessel Head penetrations Doel 1

Status in June 2017

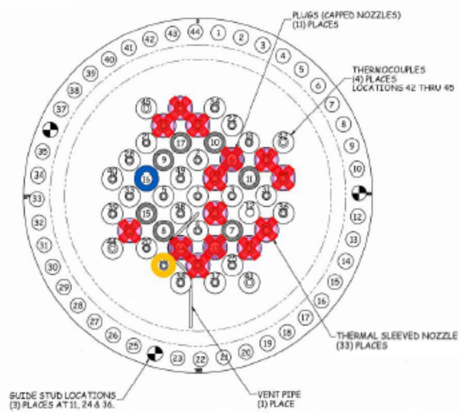
- 49 adapters
- VT: no degradation reported
- UT: 25 relevant indications in 14 adapters
  - OD surface indications, considered axial, close to J-groove weld fusion line
  - Since 2016, 2 new indications and slight growth of known indications (but << to predictions)
  - Justification for Continued Operation up to outage June 2018
  - Qualification of Inside Diameter Temper Bead repair in 2017
  - June 2018: ISI of all 49 adapters, and repair of 14 affected adapters

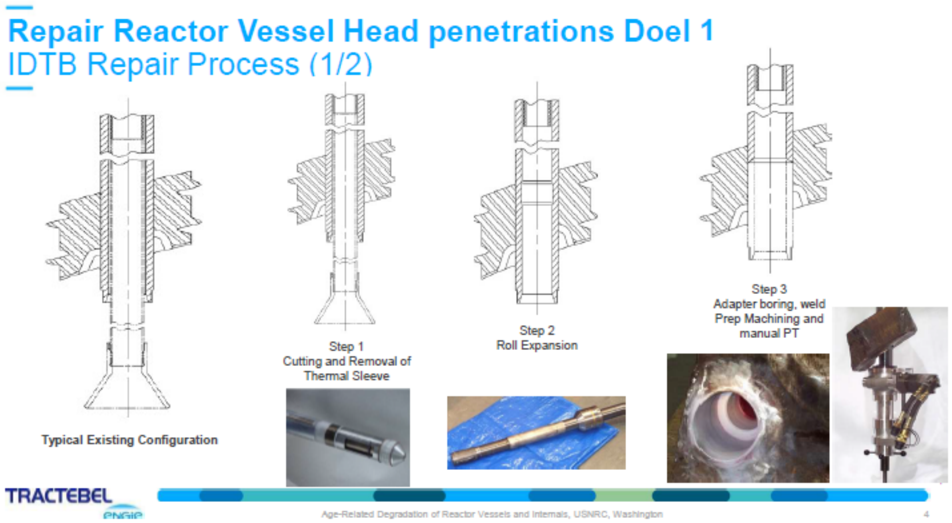


Repair Reactor Vessel Head penetrations Doel 1

Inspection and repair in 2018

- Doel 1 – 2018: UT + repair by IDTB
  - Pre-ISI PWSCC locations ◎
  - Originally scheduled (14) repair locations ✖
  - Two new indications detected by UT in 2018
    - Penetration 26 : seen not reported in 2017 ○
    - Penetration 16 : not observed in 2017 ○
  - Total of 16 repaired penetrations





Repair Reactor Vessel Head penetrations Doel 1

IDTB Repair Process (1/2)

Typical Existing Configuration

Step 1
Cutting and Removal of Thermal Sleeve

Step 2
Roll Expansion

Step 3
Adapter boring, weld Prep Machining and manual PT

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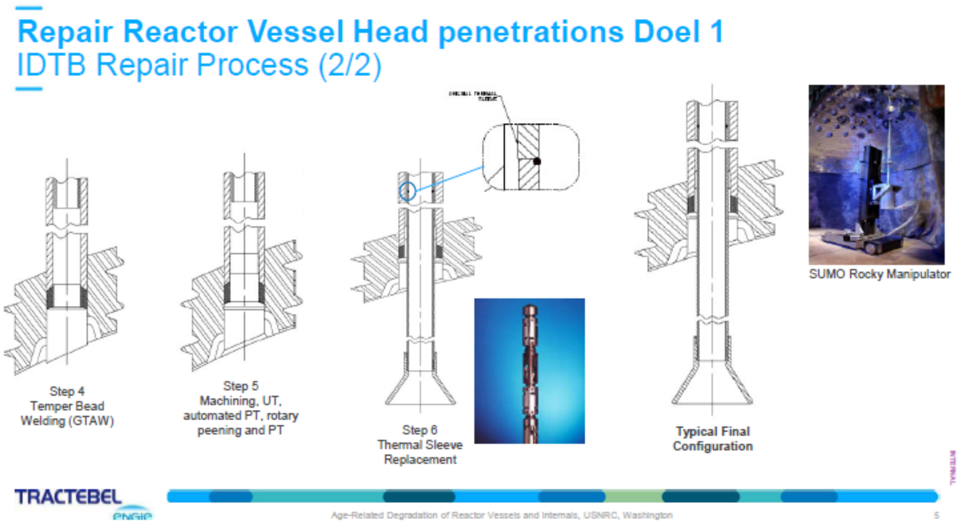

Repair Reactor Vessel Head penetrations Doel 1

IDTB Repair Process (2/2)

Step 4
Temper Bead Welding (GTAW)

Step 5
Machining, UT, automated PT, rotary peening and PT

Step 6
Thermal Sleeve Replacement

Typical Final Configuration

SUMO Rocky Manipulator

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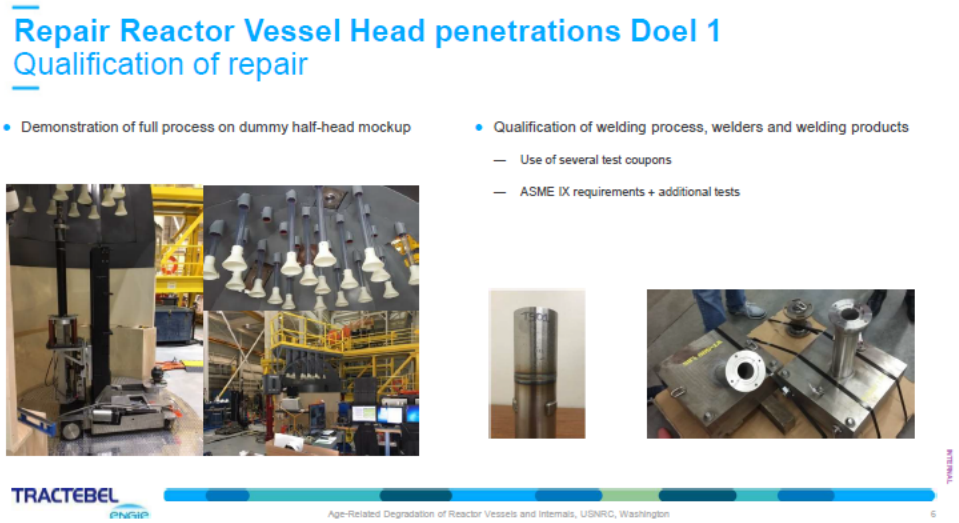
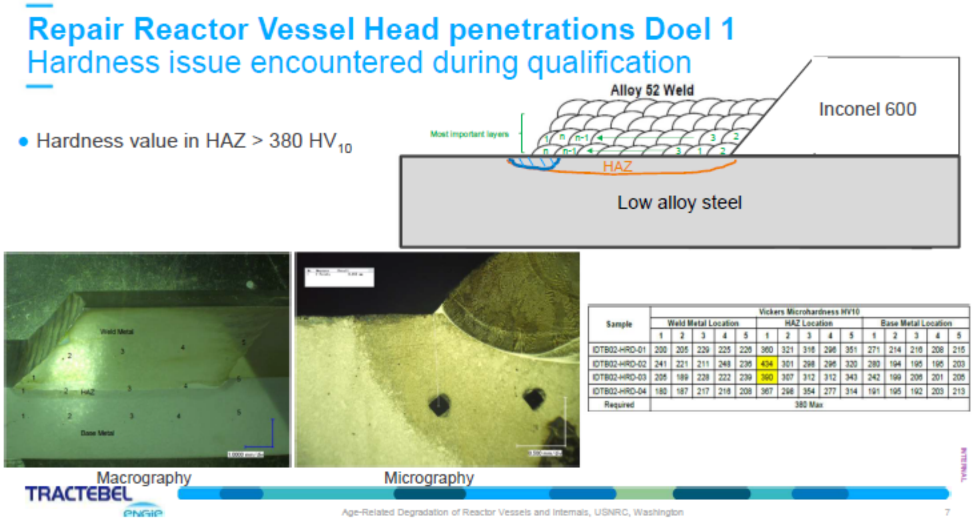


Repair Reactor Vessel Head penetrations Doel 1
Qualification of repair

- Demonstration of full process on dummy half-head mockup

- Qualification of welding process, welders and welding products
  - Use of several test coupons
  - ASME IX requirements + additional tests



Repair Reactor Vessel Head penetrations Doel 1
Hardness issue encountered during qualification

- Hardness value in HAZ > 380 HV_{10}

Sample	Vickers Microhardness HV10														
	Weld Metal Location					HAZ Location					Base Metal Location				
	1	2	3	4	5	1	2	3	4	5	1	2	3	4	5
IDTB02-HRD-01	200	205	229	225	226	360	321	316	296	351	271	214	216	208	215
IDTB02-HRD-02	241	221	211	248	236	434	301	298	296	320	280	194	195	195	203
IDTB02-HRD-03	205	189	228	222	239	390	307	312	312	343	242	199	206	201	205
IDTB02-HRD-04	180	187	217	216	208	367	298	354	277	314	191	195	192	203	213
Required	380 Max														

Repair Reactor Vessel Head penetrations Doel 1

Issues encountered during repair

- During qualification, welding parameters and sequence had to be optimized in order to have sufficient tempering and acceptable hardness values in HAZ
- During repair, breakdown of cutting tools used to cut and remove thermal sleeves
  - FRAMATOME Inc. never encountered this issue before
  - Issue may have been due to a very slight inclination of the thermal sleeves (approx. 0.5-1°)
  - Solution : tooling adaptations with tests done at Lynchburg and copied at Doel

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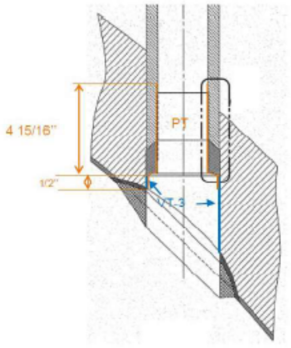
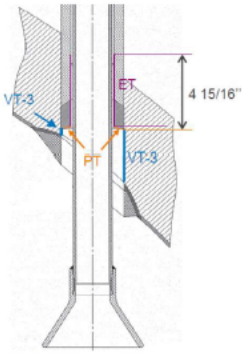
In-Service-Inspection of repaired penetrations

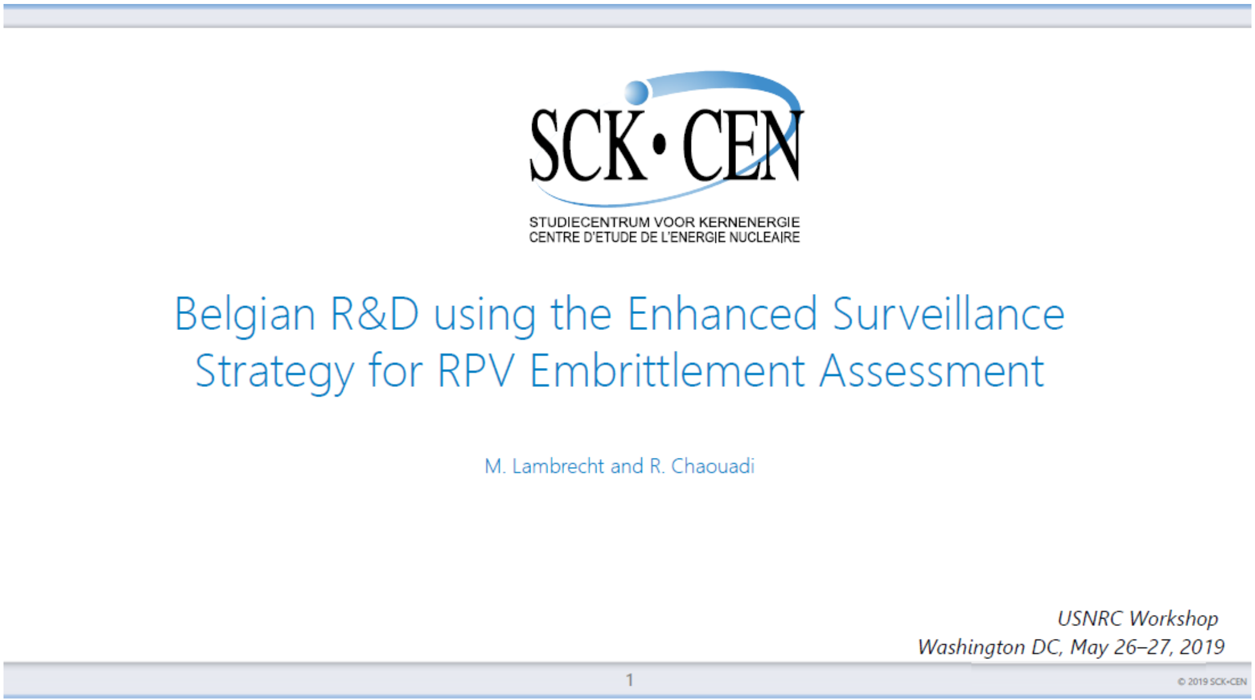
- Without thermal sleeve



- With thermal sleeve



B.12 Belgian R&D Using the Enhanced Surveillance Strategy for RPV Embrittlement Assessment (M. Lambrecht)



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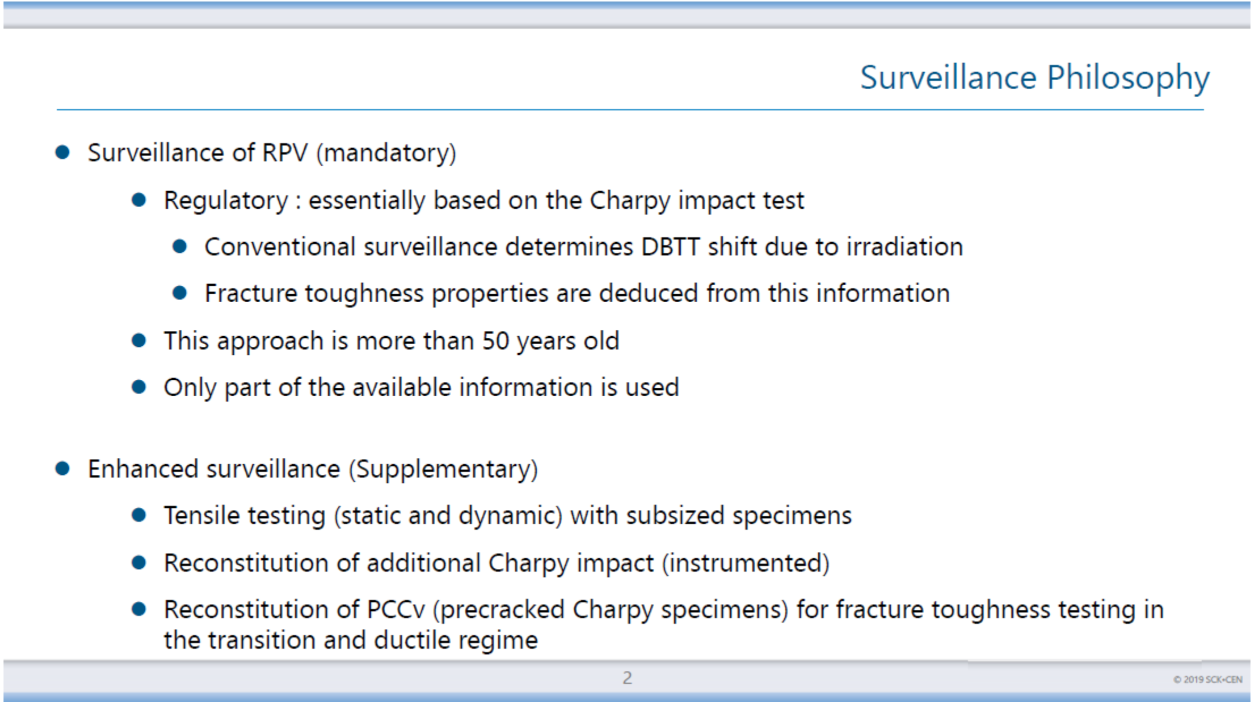
Belgian R&D using the Enhanced Surveillance Strategy for RPV Embrittlement Assessment

M. Lambrecht and R. Chaouadi

USNRC Workshop
Washington DC, May 26–27, 2019



Surveillance Philosophy

- Surveillance of RPV (mandatory)
  - Regulatory : essentially based on the Charpy impact test
    - Conventional surveillance determines DBTT shift due to irradiation
    - Fracture toughness properties are deduced from this information
  - This approach is more than 50 years old
  - Only part of the available information is used

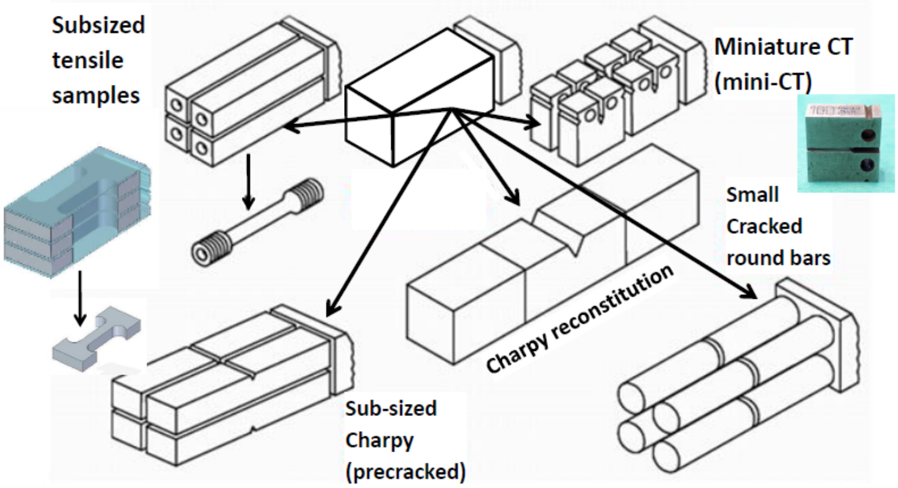
- Enhanced surveillance (Supplementary)
  - Tensile testing (static and dynamic) with subsized specimens
  - Reconstitution of additional Charpy impact (instrumented)
  - Reconstitution of PCCv (precracked Charpy specimens) for fracture toughness testing in the transition and ductile regime

PART 1:

Reconstitution

What can be done with a broken Charpy impact specimen?

Specimen Reconstitution and Miniaturization



Subsized tensile samples

Miniature CT (mini-CT)

Small Cracked round bars

Charpy reconstitution

Sub-sized Charpy (precracked)

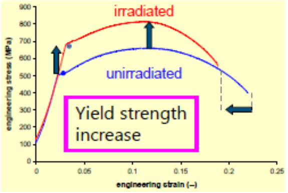
PART 2:

Tensile and Fracture toughness testing

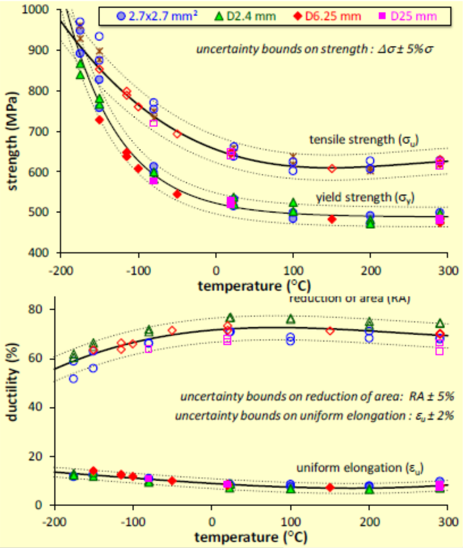
More than Charpy impact

Tensile testing

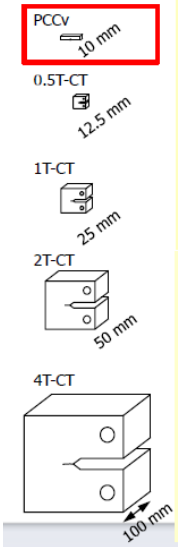
Tensile Testing

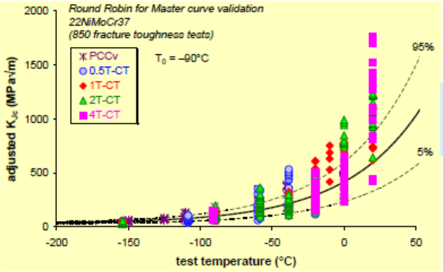


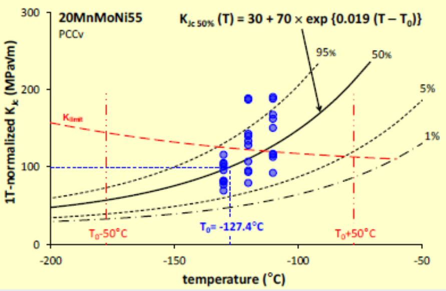
➔ Radiation Damage Modeling



Master Curve and Crack Resistance Curve







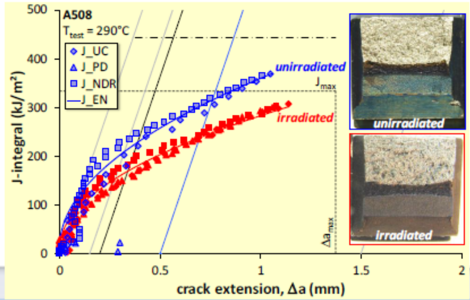
→ ASTM E1921

$K_{J\,med}$ = 30 + 70 exp [0.019 (T – $\mathbf{T_0}$)]

Fracture toughness transition curve characterized by one single parameter, $\mathbf{T_0}$

Consistency

UC : Unloading Compliance
PD : Potential Drop
NDR : Normalization Data Reduction
EN : Energy Normalization

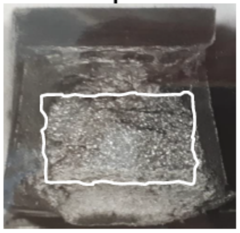


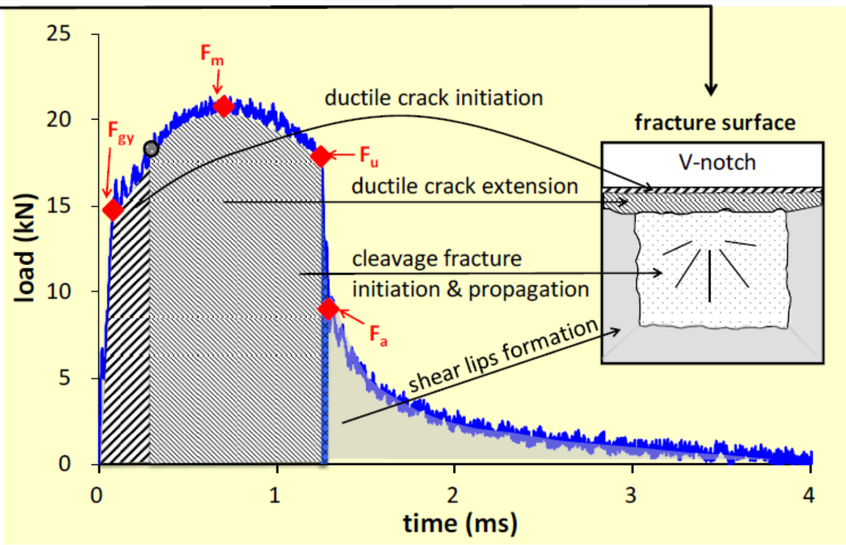
PART 3:

All What You Can Extract from the Instrumented Charpy Impact Test

Everything you always wanted to know about the Charpy impact test

Instrumented Charpy Impact Test : Load Diagram

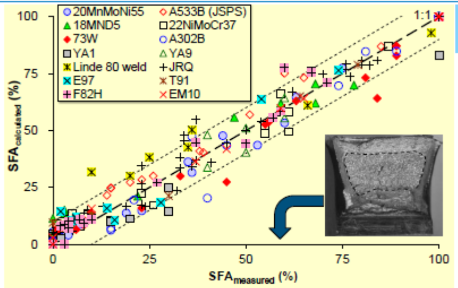




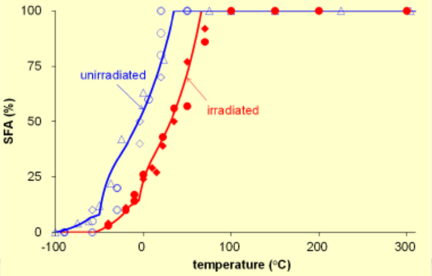
Characteristic loads

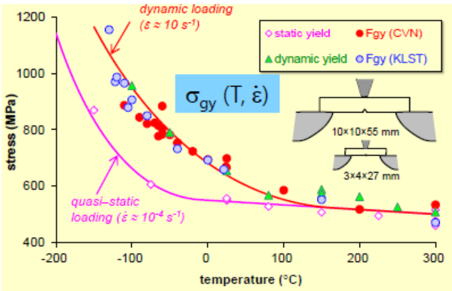
- **F_{gy}** (general yield load)
- **F_m** (maximum load)
- **F_u** (unstable fracture load)
- **F_a** (arrest load)

Parameters from instrumented Charpy curve : SFA & tensile yield



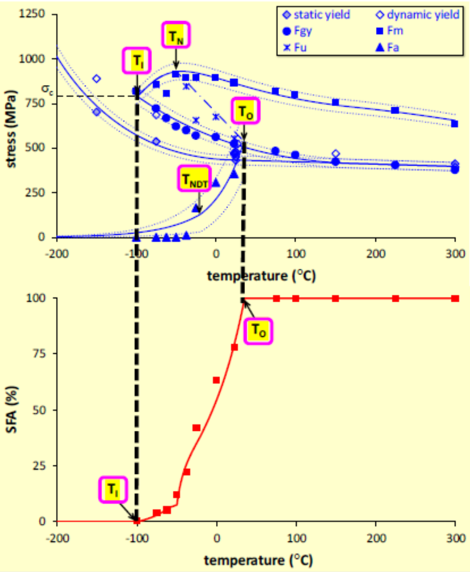
$SFA_{calculated} = f(F_{gy}, F_m, F_u, F_a)$ ①





② Unique yield strength function σ_y (T, $\dot{\varepsilon}$)

Load Diagram : Characteristic temperatures



T_I : brittle fracture temperature (SFA=0%)

T_O : ductile fracture temperature (SFA=100%)

$T \leq T_I \rightarrow$ SFA = 0%

$T_I < T < T_O \rightarrow 0\% <$ SFA $< 100\%$

$T \geq T_O \rightarrow$ SFA = 100%

➔ $\sigma(T_I) \rightarrow$ microcleavage fracture stress (σ_c)

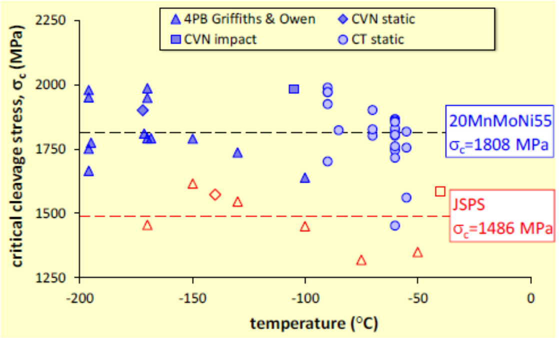
➔ T_{NDT} (based on arrest loads) correlation to Pellini NDT

➔ T_I (CVN) correlation to $T_{100MPa\sqrt{m}}$ (static fracture toughness) correlation

---

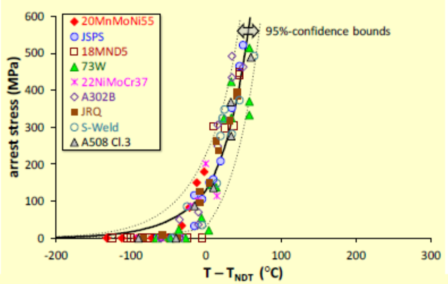
Parameters from instrumented Charpy curve: Cleavage Fracture Stress & T_{NDT}

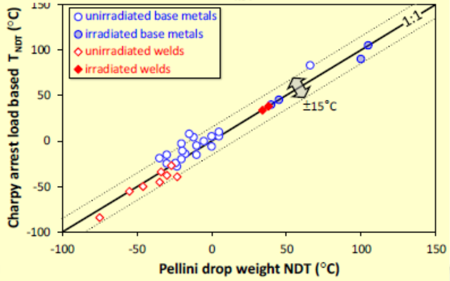
③ $\sigma_c = K_p \times \sigma_{gy}(T_I)$



④ $T_{NDT} = f(F_a)$

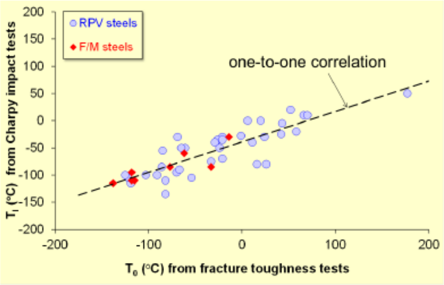
$T_{NDT\ (Charpy)}$–$NDT_{Pellini}$ correlation

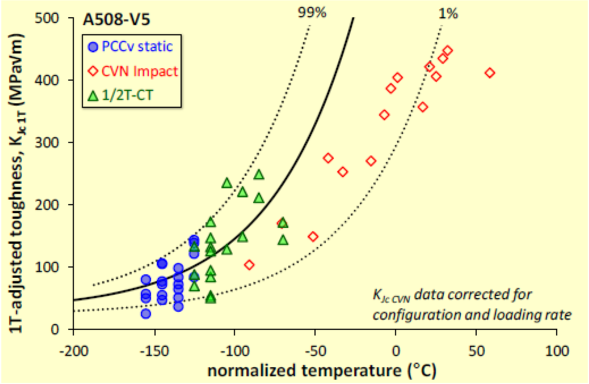




Parameters from instrumented Charpy curve : Master Curve

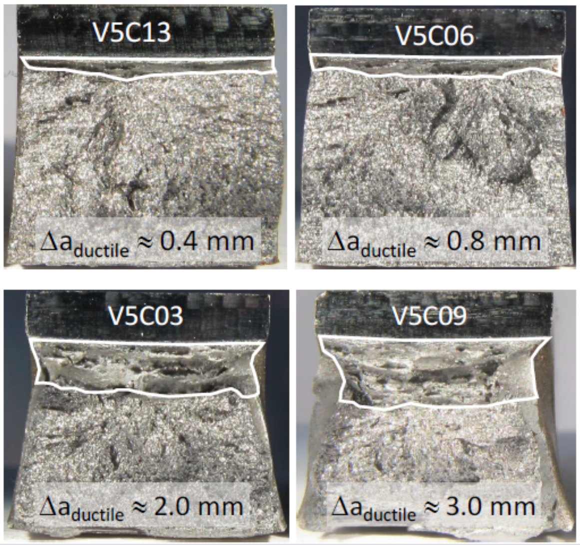
⑤ $T_{I\ (Charpy\ impact)} - T_{0\ (fracture\ toughness)}$ correlation



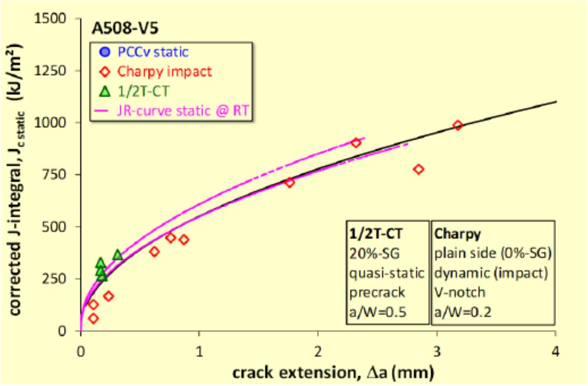


Correction for difference in geometry and test conditions. The correction factors are established once for all and are not modified to fit the data.

Parameters from instrumented Charpy curve : J_R–Curve

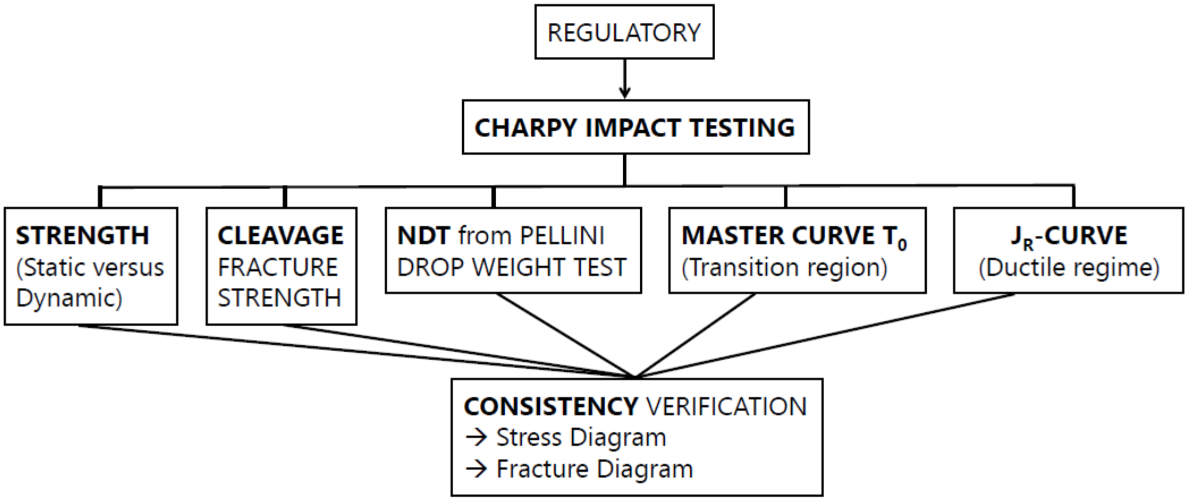


⑥ J–Δa curve



Correction of $J_R^{CVN\ impact}$ to obtain J_R^{static} accounting for effects of geometry and loading conditions

The Central Role of the Charpy Impact Test



REGULATORY

CHARPY IMPACT TESTING

STRENGTH (Static versus Dynamic)

CLEAVAGE FRACTURE STRENGTH

NDT from PELLINI DROP WEIGHT TEST

MASTER CURVE T_0 (Transition region)

J_R-CURVE (Ductile regime)

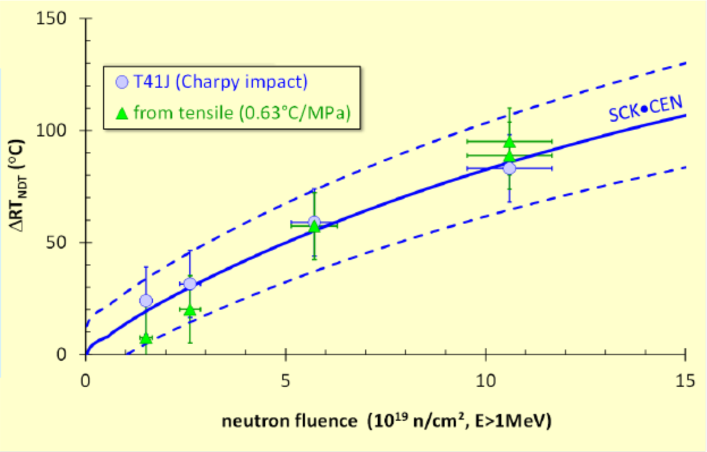
CONSISTENCY VERIFICATION
→ Stress Diagram
→ Fracture Diagram

Summary and Conclusions : RPV Surveillance : Belgian Practice

- Surveillance programs of the Belgian reactors → two reports per surveillance capsule :
  - First report : **Regulatory** surveillance based on the RT_{NDT} concept (typically, Charpy impact tests and a few tensile tests)
  - Second report : Enhanced surveillance → **enhanced reliability**
    - Additional tensile tests in a wide range of test temperature, sometimes at other strain rates
    - Specimen reconstitution (precracked Charpy specimens) for fracture toughness tests in the transition and at upper shelf temperature (operation temperature)
    - Radiation damage modeling and embrittlement trend curves
- **Property-to-property correlation** (tensile, Charpy impact and fracture toughness) and **modeling**
- **Increased reliability** in the surveillance database in the perspective of **long term operation** (LTO)
- Database supported by MTR data (mainly BR2)

Embrittlement Trend Curve

- ➔ Physically-based embrittlement trend curve accounting for material variables (Cu, Ni, P) and irradiation variables (ϕ, Φ, T_{irrad})
- ➔ Continuous development and improvement of the model with the support of MTR BR2 data and reliable databases



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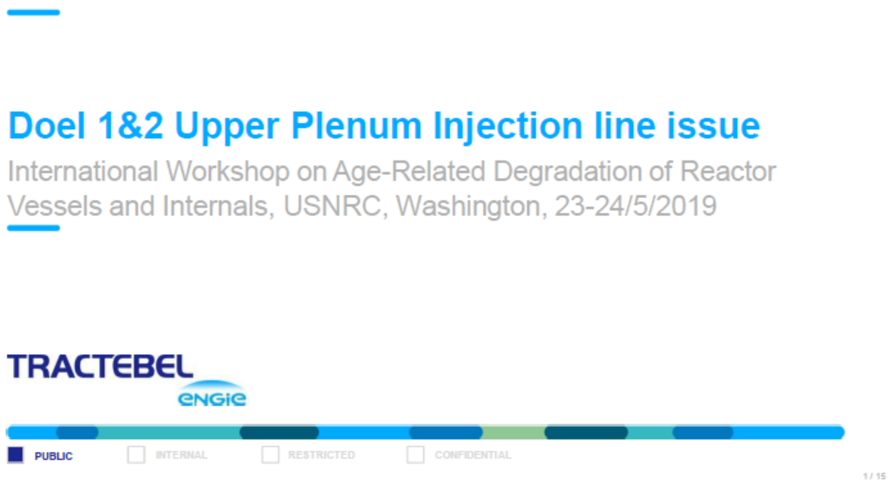
B.13 Doel 1 & 2 Upper Plenum Injection Line Issue (M. De Smet)



Doel 1&2 Upper Plenum Injection line issue

International Workshop on Age-Related Degradation of Reactor Vessels and Internals, USNRC, Washington, 23-24/5/2019

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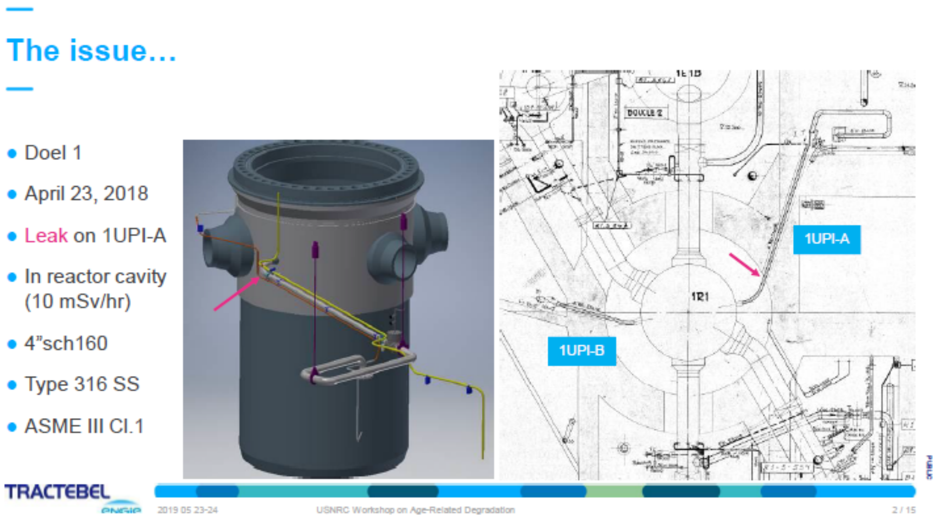

The issue…

- Doel 1
- April 23, 2018
- Leak on 1UPI-A
- In reactor cavity (10 mSv/hr)
- 4"sch160
- Type 316 SS
- ASME III Cl.1

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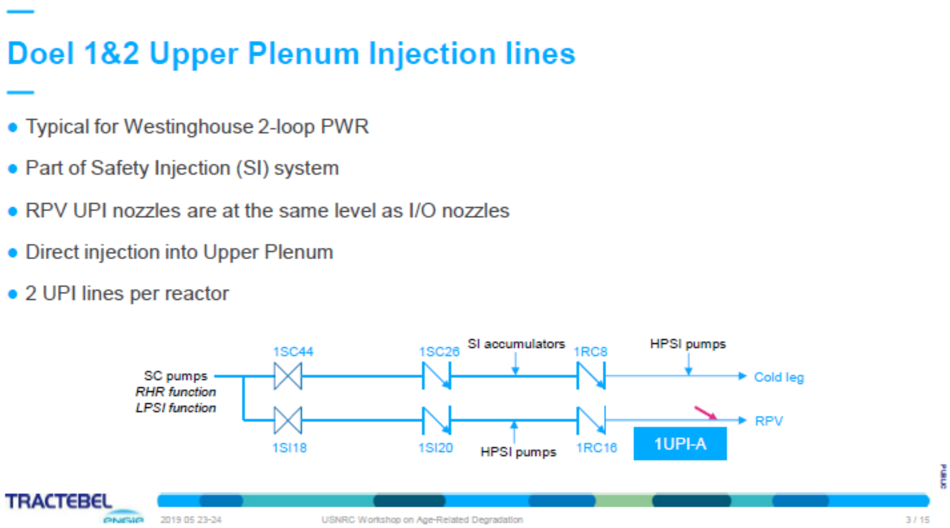

Doel 1&2 Upper Plenum Injection lines

- Typical for Westinghouse 2-loop PWR
- Part of Safety Injection (SI) system
- RPV UPI nozzles are at the same level as I/O nozzles
- Direct injection into Upper Plenum
- 2 UPI lines per reactor

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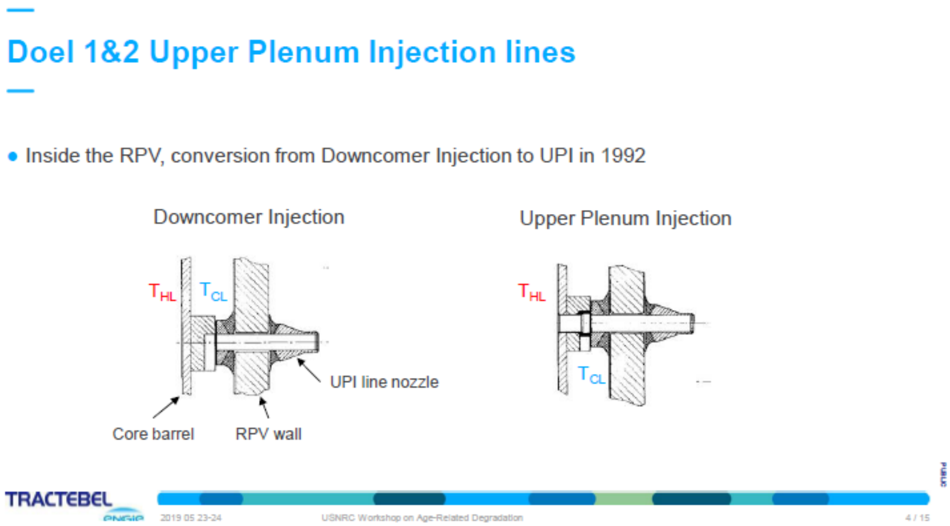

Doel 1&2 Upper Plenum Injection lines

- Inside the RPV, conversion from Downcomer Injection to UPI in 1992

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Non-destructive examinations

- Mechanized VT, ET, UT, UT-TOFD (FRAMATOME GmbH Erlangen) of all 4 UPI lines
  - From inside, access through check-valve
  - Dry (pneumatic plug in RPV nozzle)
  - Several carriers
- Manual UT of welds
  - From outside

Destructive examinations

- Doel 1 UPI-A ► SCK•CEN (Mol, Belgium) ►► THERMAL FATIGUE
- Doel 2 UPI-A ► FRAMATOME GmbH (Erlangen, Germany) ►► THERMAL FATIGUE

Overview observed degradation

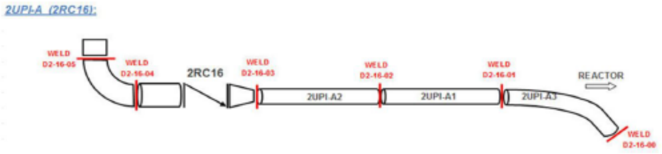
	UPI-A straight pipe	UPI-A weld	UPI-A elbow
Doel 1	Cracks Leak	Circumferential crack	-
Doel 2	Cracks	Circumferential crack	-

Susceptibility for fatigue

- Screening of non-isolable RCS branches, including UPI lines (RPV branches), in the framework of Periodic Safety Review
- No in-leakage ►► no inspection necessary ?
  (*EPRI MRP-146 Rev2 excludes Horizontal RCS branch lines without in-leakage from further evaluation*)
- UPI-A considered to be more susceptible than UPI-B based on its:
  - Length (8m, versus 4m for UPI-B)
  - Slope (upward to RPV)
- Recommended inspection location difficult to access and with high dosimetry

Repair

- Affected straight pipes of Doel 1&2 A-lines were replaced by new A2 and A1 parts



2UPI-A (2RC16):

- Challenging environmental conditions: radiation level and limited space, difficult to access
  - Shielding
  - Training on full-scale mock-up

Repair – Mock-up

Justification for safe restart – Defence-In-Depth principle

Level 1

- Repaired UPI lines are free of indications
- Degradation is confirmed to be due to fatigue
- Validation of phenomenon at the origin of fatigue

Level 2

- Monitoring of UPI lines
- Future inspection programme of UPI lines to timely detect damage

Level 3

- Definition of conservative undetected flaw in UPI line
- Stability of postulated flaw under all possible operating conditions
- Management of UPI line ruptures

Monitoring & future inspections

- Monitoring of 4 UPI lines
  - Temperatures (FAMOSi)
  - Displacements (LABORELEC)
  - Accelerations (LABORELEC)
- Inspection of 4 UPI lines at next outage
  - Base metal
  - Welds
  - Combination of VT, ET, UT, RT

First observations from monitoring

MS09

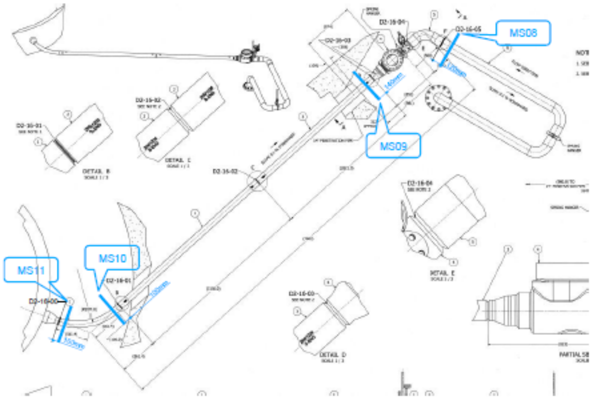
$\Delta T_{\text{top-bottom}}$ ~40°C (A,B)
Range: 120°C – 160°C (A)
Range: 200°C – 240°C (B)

MS10

$\Delta T_{\text{top-bottom}}$ <10°C (A)
$\Delta T_{\text{top-bottom}}$ <30°C (B)
Fluctuations in range: 210°C – 250°C

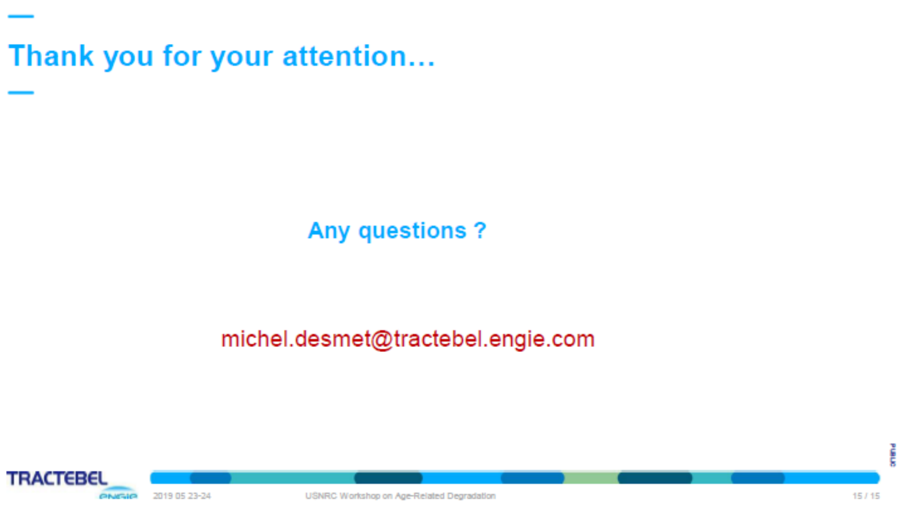
MS11

$\Delta T_{\text{top-bottom}}$~0°C
Small fluctuations as a function of time



Summary

- A leak occurred in the UPI-A line of Doel 1 in April 2018
- NDE revealed degradation in the UPI-A lines of Doel 1&2:
  - Cracking in the bottom part of the straight pipe, upstream of the weld between straight pipe and elbow
  - Circumferential cracking in the weld between straight pipe and elbow
  - No cracking was found in the UPI-B lines.
- Destructive examination confirmed degradation was due to thermal fatigue
- The straight parts of the UPI-A lines were replaced
- Safety demonstration relies on repair, monitoring and future inspections
- Monitoring confirms the presence of thermal cycles in UPI lines at full power
- Structural integrity evaluations (stress, fatigue, fatigue crack growth) are on-going



Thank you for your attention…

Any questions ?

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B.14 Inspection of Control Rod Guide Assemblies in Belgian NPPs (C. Dupuit)

Inspection of Control Rod Guide Assemblies in Belgian NPPs

International Workshop on Age-Related Degradation of Reactor Vessels and Internals, USNRC, Washington, 23-24/5/2019

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23/05/2019 Age-Related Degradation of Reactor Vessels and Internals, USNRC, Washington 1

Background

- International guidelines and recommendations
  - EPRI report MRP-227-A (2011): 'PWR Internals Inspection and Evaluation Guidelines'
  - Westinghouse NSAL-10-1 (2010): RCCA Guide Card Wear -> Lifetime expectations
  - Westinghouse WCAP-17451-P (2013): Westinghouse Fleet Operational Projections
    -> Effective Full Power Years (EFPY) for baseline inspections

GT design	Units	Minimum operational EFPY (85% slot opening) – NSAL-10-1	EFPY for baseline inspections in WCAP-17451-P
14x14	Doel 1-2	> 40	38 to 42
15x15	Tihange 1	36	40 to 44
17x17 AS 17x17 AXLR	Tihange 2 / Doel 3 Tihange 3 / Doel 4	>40 >40	34 to 38 32 to 36

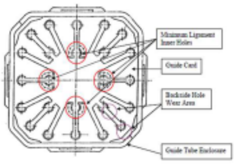
  - Westinghouse NSAL-17-1 (2017): Recent operating experience with wear measurements in plants with 17x17 A or 17x17 AS guide tubes operating with ion-nitride RCCAs:
    -> Higher wear rate than expected -> Criteria defined in WCAP-17451-P are potentially non-conservative

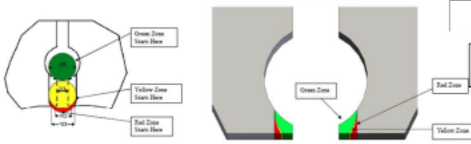
Wear criteria

- GCWM inspection decided after NSAL 17-1
- Wear criteria

Guide Card Slot Opening Criteria Widths

W_1 = Nominal width of unworn guide card slot
W_2 = 0,8 x RCCA rodlet diameter (DR)
W_3 = 0,85 x RCCA rodlet diameter (DR)





Cards

Criteria Zone	Range
Green Zone	Measured slot width less than W_3 at all guide cards
Yellow Zone	Measured slot width equal to or greater than W_3 for one to (n – 1) consecutive guide card holes
Red Zone	Measured slot width equal to or greater than W_3 for (n) consecutive guide card holes

-> Tihange 3/Doel 4: Six consecutive open cards (n = 6) is an acceptable criterion

Continuous Section

Criteria Zone	Zone Range[1]
Green Zone	Ligament wear less than W_2
Yellow Zone	Ligament wear equal to or greater than W_2 and less than W_3
Red Zone	Ligament wear equal to or greater than W_3

Tihange 3 GT and RCCA characteristics

- The guide tubes are of the 17x17 AXLR design (14 ft) and have 11 guide cards and a continuous guiding zone.
- The total number of RCCAs in Tihange 3 is 52.
  - 16 ion-nitride RCCAs were loaded for the first time at the beginning of cycle 10 (1995).
  - 36 more ion-nitride RCCAs loaded for the first time at the beginning of cycle 11 (1996).
- Between 19 and 20.1 EFPY of operation with ion-nitride rods (in 2018)
- There are no non ion-nitride RCCAs remaining in the core since 1996, with some exemptions since 2010 where a few non ion-nitride RCCAs were used occasionally

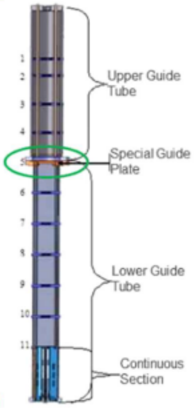
Tihange 3 GCWM results

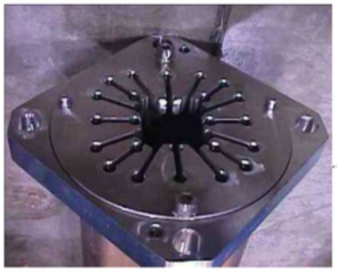
- GCWM inspection performed in 04/2018
- Extensive guide cards wear:
  - 35 guide tubes in "green" zone
  - 8 guide tubes in "yellow" zone
  - 9 guide tubes in "red" zone

    -> 15 guide tubes were predicted not to be able to make two more fuel cycles ➔ contingency measures

    - 2 guide tubes swapped with spare tubes already in reactor
    - 13 "Special Guide Plates" (specific feature of 17x17 guide tubes, where one guide plate is bolted and can be removed) replaced by Thicker Special Guide Plates (1.5 inch thick instead of 1 inch)

Tihange 3 Thicker Special Guide Plates



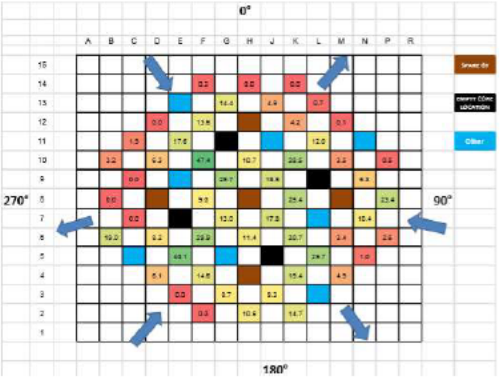


Special Guide Plate (SGP)

Thicker Special Guide Plate (TSGP)

Tihange 3 GCWM results

- Core map with #EFPY to Red Zone:



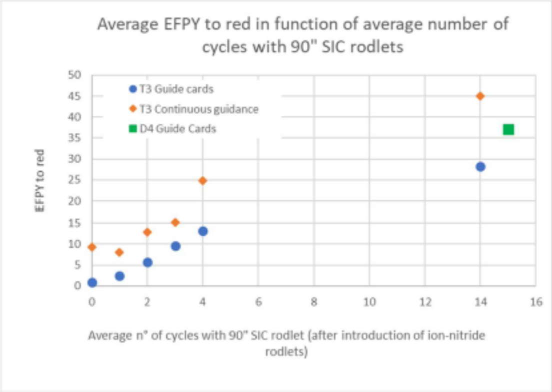
Doel 4

- Same design as Tihange 3
- All RCCAs are ion-nitride since cycle 15 (1999).
- 18.2 EFPY of operation with ion-nitride rods (in 2018)
- Following Tihange 3 inspection in 04/2018 significant wear was also expected in Doel 4.
- GCWM performed in 11/2018
- Surprisingly the results were very good, all guide tubes in green “condition”, minimum 12.7 EFPY before reaching “red” zone for most worn tube.

Possible reason of differences between Doel 4 and Tihange 3

- Internal analysis: Both units use “hybrid” RCCAs with combination of Ag-In-Cd in lower part and B_4C above; there is however one difference:
  - In Doel 4 all RCCAs have 90” Ag-In-Cd (two bars of 50” and 40”) since ion-nitride rods were introduced
  - In Tihange 3 most RCCAs were 40” Ag-In-Cd from 1996 to 2012, except for D and SD banks (where RCCAs with 90” Ag-In-Cd were used).
  - Since 2012, 90” Ag-In-Cd RCCAs are progressively introduced in all positions
  - Guide tubes of D and SD banks in Tihange 3 show much less wear (“green”)
  - 90” Ag-In-Cd RCCAs are heavier and stiffer, which influences vibration behaviour
  - There is a correlation between the number of cycles operated with 90” Ag-In-Cd RCCAs and the average number of EFPY to “red” condition, although the scatter is very high
- Westinghouse has been requested to come up with a reasoning – on-going

Possible reason of differences between Doel 4 and Tihange 3



Doel 1&2

- The 33 guide tubes are of the 14x14 design (8 ft). There are 7 guide cards and a continuous guiding zone.
- RCCAs with chrome-plated rodlets were introduced in 2005 to 2007 in Doel 1
  - In 2016, 7 of the 8 RCCAs introduced in 2005 were replaced by old standard RCCAs
- 20 RCCAs with chrome-plated rodlets were introduced in 2010-2011 in Doel 2, the remaining 13 are standard stainless steel rodlets.
- The guide cards and continuous section were inspected in all 33 guide tubes in Doel 1 and in Doel 2 in 06/2018 and only very limited wear was measured. All in “green” status
- Wear projections conclude it will be > 100 EFPY before any guide card or continuous section is projected to enter the red zone.

Doel 3 & Tihange 2

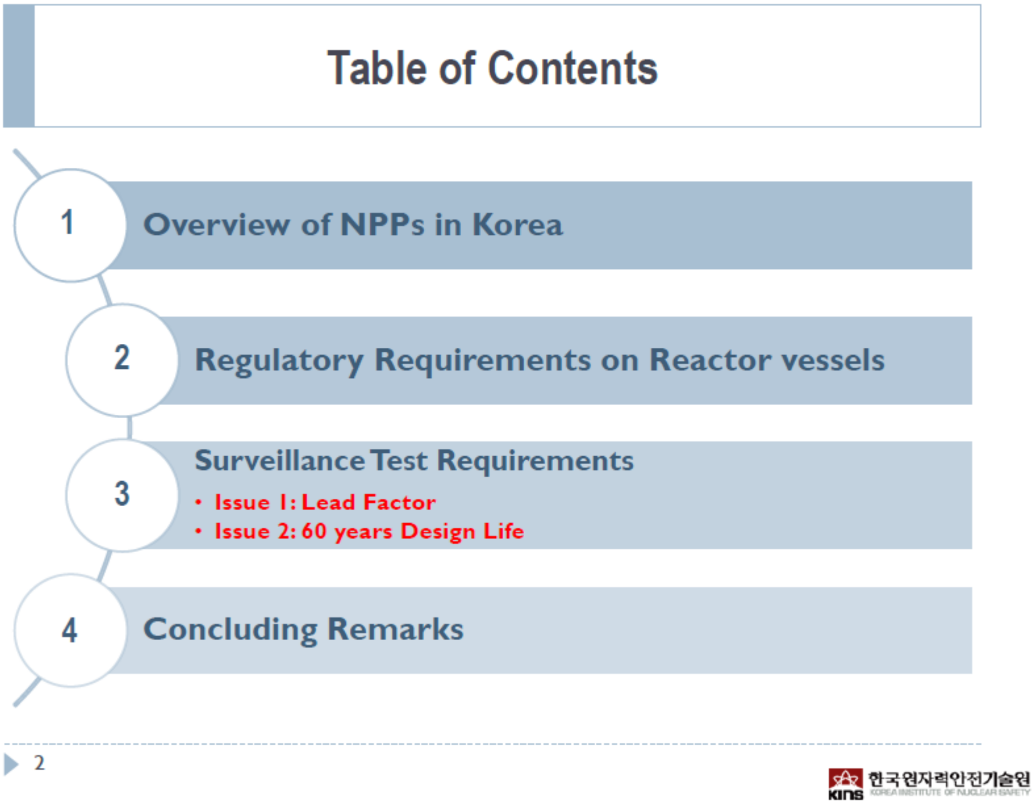
- Guide tube of 17 x 17 AS design (12 ft, 10 guide cards and shorter continuous section than 17 x 17 standard)
- Ion-nitride RCCAs introduced between 1994 and 1999, all ion-nitride since 1999
- RCCAs are Ag-In-Cd type
- Based on WCAP-17451 Rev.1, GT had more than 34 EFPY before reaching “red” zone
- In 2018 Westinghouse reviewed FME videos from split pins replacement (2006 in D3, 2001 in T2)
- No significant wear identified, wear projection conclude to a time of 42.3 EFPY for next inspection
- This is beyond final shutdown date of 2022/2023.

B.15 Current Status of Aging Management on Reactor Vessels in Korea (focusing on surveillance test) (T-K Song)

International Workshop on Age-Related Degradation of Reactor Vessels and Internals
23-24 May 2019, NRC, USA

Current Status of Aging Management on Reactor Vessels in Korea (focusing on surveillance test)

Tae-Kwang Song, Yong-Beum Kim
KINS



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한국원자력안전기술원 KOREA INSTITUTE OF NUCLEAR SAFETY

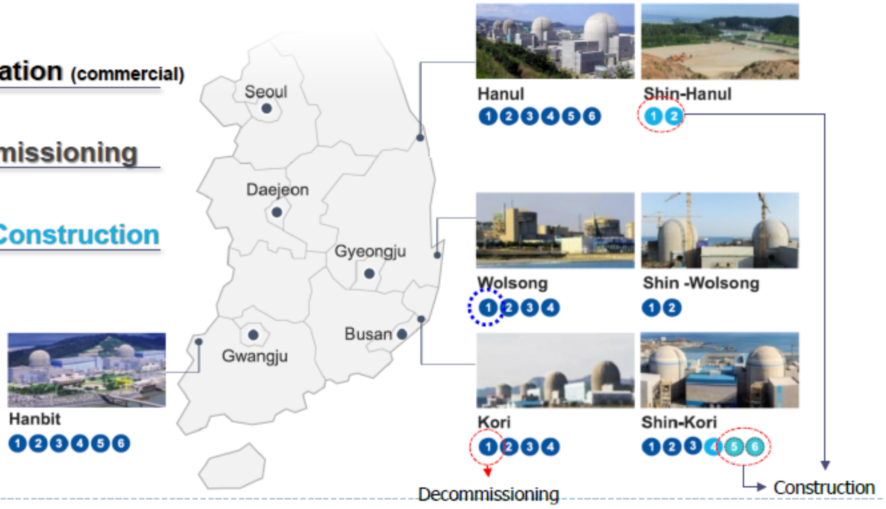
Overview of NPPs in Korea (1)

- Status of Nuclear Power Plants in Korea
  - As of May 2019

In Operation (commercial)
24 Units

Decommissioning
1 Unit

Under Construction
5 Units



Overview of NPPs in Korea (2)

- Current Status of Operating NPPs

Site	Unit	MW	Reactor Type	Commercial Operation
Kori	2	650	Westinghouse	July 1983
	3	950	Westinghouse	Sep. 1985
	4	950	Westinghouse	April 1986
Shin-Kori	1	1000	OPR-1000	April 2011
	2	1000	OPR-1000	July 2012
	3	1400	APR-1400	Dec. 2016
Wolsong	1	679	PHWR	April 1983
	2	700	PHWR	July 1997
	3	700	PHWR	July 1998
	4	700	PHWR	Oct. 1999
Shin-Wol song	1	1000	OPR-1000	July 2012
	2	1000	OPR-1000	July 2015
Hanbit	1	950	Westinghouse	Aug. 1986
	2	950	Westinghouse	June 1987
	3	1000	OPR-1000	Mar. 1995
	4	1000	OPR-1000	Jan. 1996
	5	1000	OPR-1000	May 2002
	6	1000	OPR-1000	Dec. 2002
Hanul	1	950	Framatome	Sep. 1988
	2	950	Framatome	Sep. 1989
	3	1000	OPR-1000	Aug 1998
	4	1000	OPR-1000	Dec 1999
	5	1000	OPR-1000	July 2004
	6	1000	OPR-1000	April 2005

Overview of NPPs in Korea (3)

- Reactor Types

"different reactor type(WH vs KSNP)"

"different design life(40 yrs vs 60 yrs)"

Westinghouse Type	• Kori Units 1~4, Hanbit* Units 1&2
Framatome Type	• Hanul** Units 1&2
KSNP**** Type (OPR-1000)	• Hanbit Units 3~6, Hanul Units 3~6 • Shin***-Kori Units 1&2, Shin-Wolsong Units 1&2
KSNP Type (APR-1400)	• Shin-Kori Units 3~6, Shin-Hanul Units 1&2
PHWR Type	• Wolsong Units 1~4

* Former name is Younggwang

** Former name is Uljin

*** The term of 'Shin-, 新' means 'new'

**** KSNP(Korea Standard Nuclear Plant) has been developed based on CE type reactor

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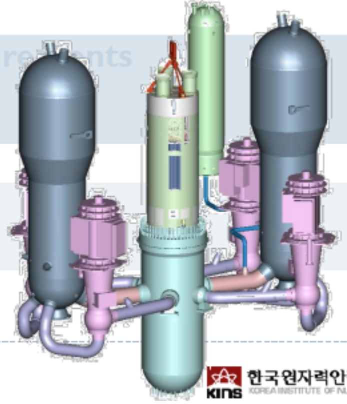
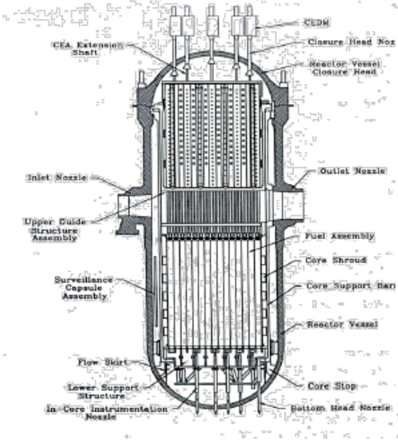
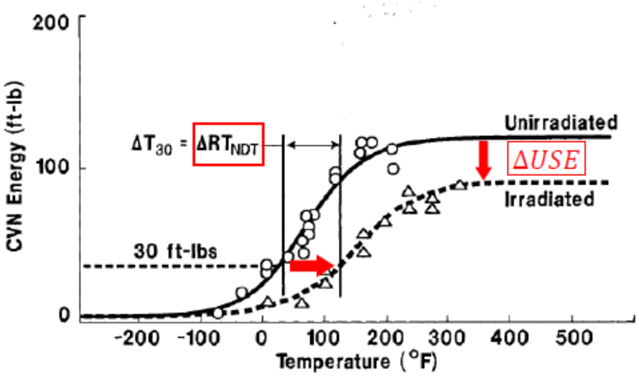


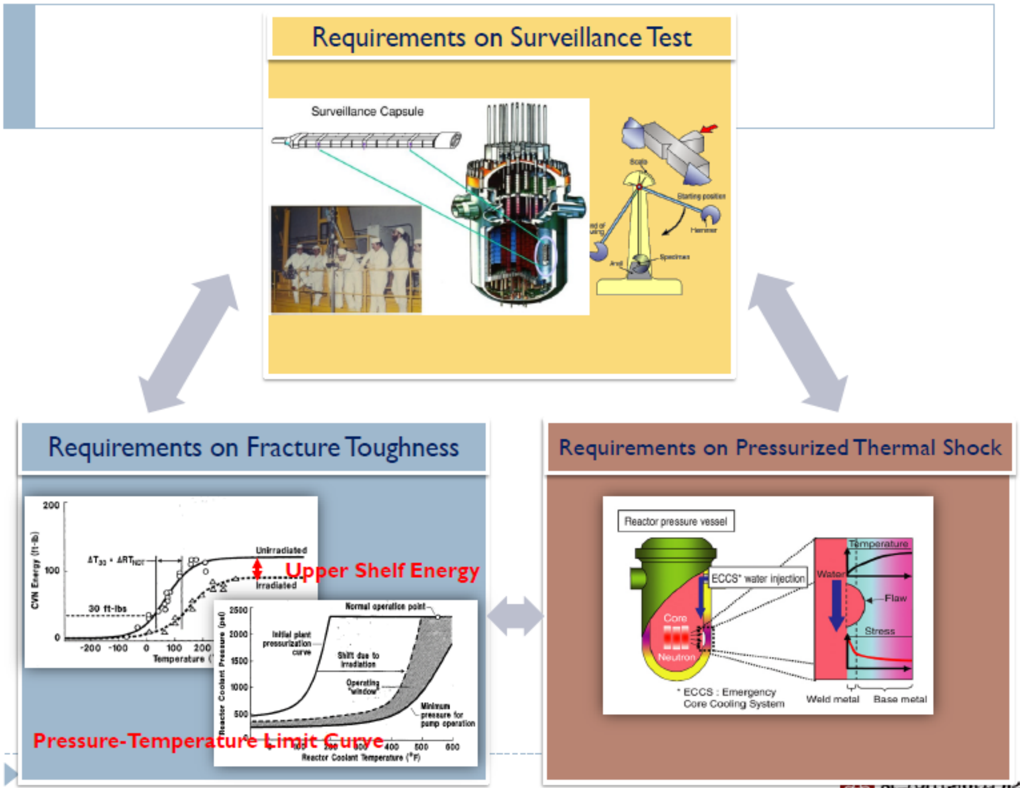
Irradiation Embrittlement

- Irradiation Embrittlement
  - If fast neutron fluence(E≥1.0MeV) exceeds 10^{17} n/cm^2, irradiation embrittlement is introduced in typical low alloy ferritic RPV material





$$Adjusted\ RT_{NDT} = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin$$

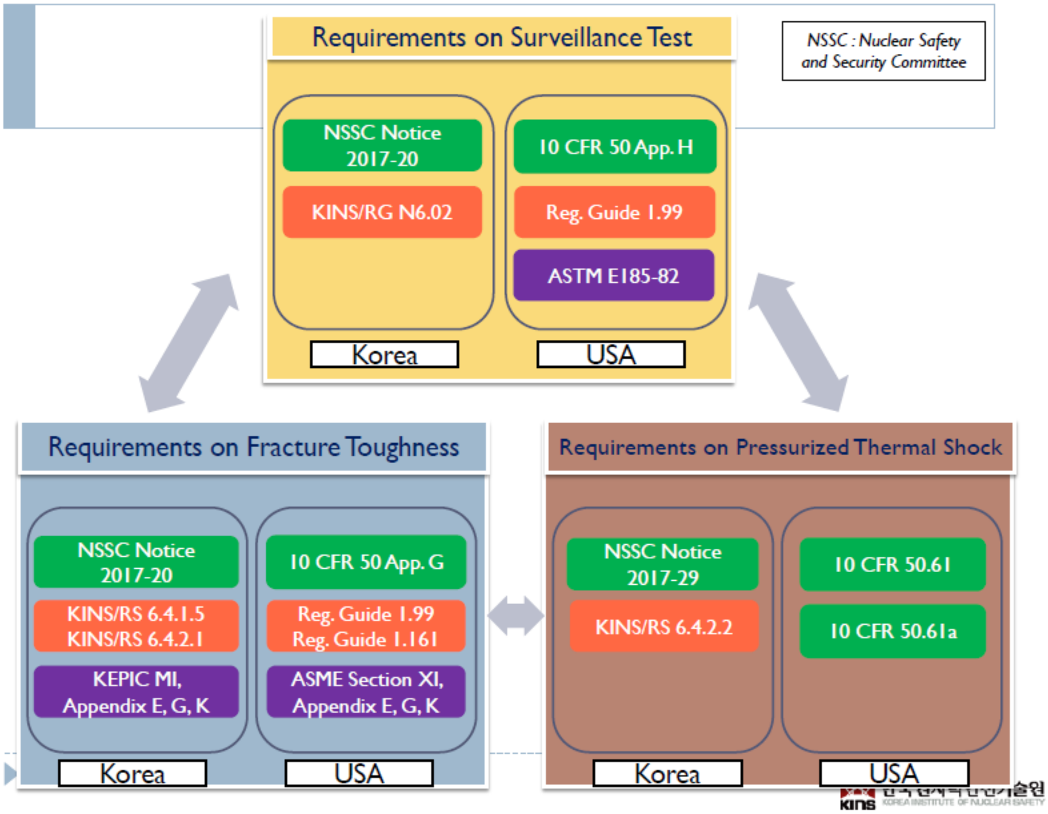


Requirements on Surveillance Test

Requirements on Fracture Toughness

Requirements on Pressurized Thermal Shock

NSSC : Nuclear Safety and Security Committee



Requirements on Surveillance Test

Korea	USA
NSSC Notice 2017-20	10 CFR 50 App. H
KINS/RG N6.02	Reg. Guide 1.99
	ASTM E185-82

Requirements on Fracture Toughness

Korea	USA
NSSC Notice 2017-20	10 CFR 50 App. G
KINS/RS 6.4.1.5 KINS/RS 6.4.2.1	Reg. Guide 1.99 Reg. Guide 1.161
KEPIC MI, Appendix E, G, K	ASME Section XI, Appendix E, G, K

Requirements on Pressurized Thermal Shock

Korea	USA
NSSC Notice 2017-29	10 CFR 50.61
KINS/RS 6.4.2.2	10 CFR 50.61a

Surveillance Test Requirements

- Surveillance Test
  - to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region which result from exposure of these materials to neutron irradiation and the thermal environment
- NSSC Notice 2017-20 and 10 CFR 50, App. H
  - require to perform surveillance program
  - based on ASTM E185-82
- ASTM E185-82 provide surveillance program including
  - surveillance materials
  - type of specimens
  - number of specimens
  - location of capsules
  - number of capsules
  - withdrawal schedule

Withdrawal Schedule in ASTM E185-82

		Predicted Transition Temperature Shift at Vessel Inside Surface		
		≤ 56℃ (≤100 ℉)	>56 ℃(≤100 ℉) ≤ 111 ℃ (≤200 ℉)	〉 111℃ (>200 ℉)
Minimum Number of Capsules		3	4	5
Withdrawal Sequence	First	6^A	3^A	1.5^A
	Second	15^B	6^C	3^D
	Third	EOL^E	15^B	6^C
	Fourth	-	EOL^E	15^B
	Fifth	-	-	EOL^E

[A] Or at the time when the accumulated neutron fluence of the capsule exceeds $5X10^{22}$ n/m^2 ($5X10^{18}$ n/cm^2), or at the time when the highest predicted ΔRT_{NDT} of all encapsuled materials is approximately 28℃(50℉), whichever comes first.

[B] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel inner wall location, whichever comes first.

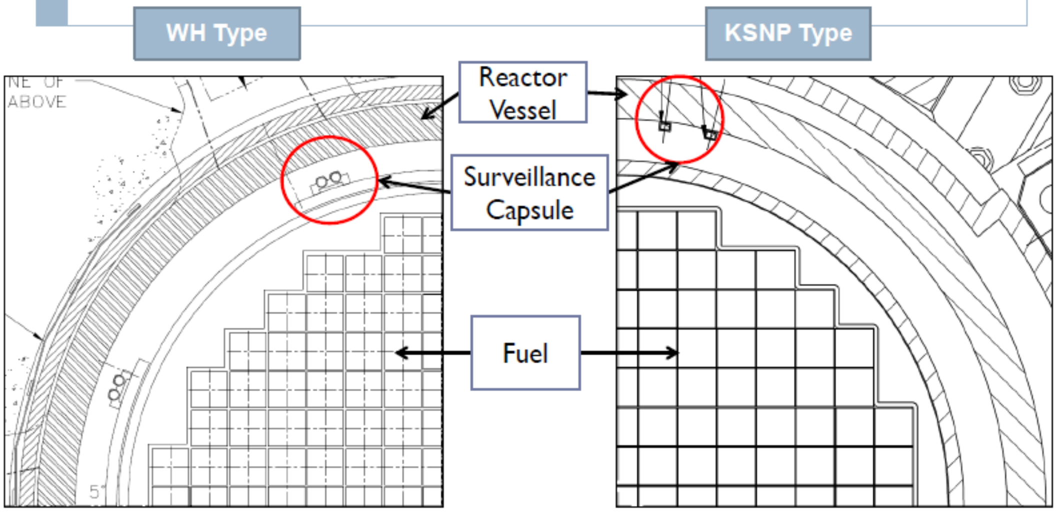
[C] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel ¼ T location, whichever comes first.

[D] Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

[E] Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.

EFPY, Effective Full Power Year

Issue 1: Lead Factor



❑ Lead Factor

- WH Type : 2.9 ~ 3.5
- KSNP Type : ~1.3

$$\text{Lead Factor} = \frac{\text{Fluence of Surveillance Capsule}}{\text{Fluence of Vessel Wall}}$$

Withdrawal Schedule of WH Type Reactor

❑ Assumption

- 40 years design life
- predicted transition shift at vessel inside surface < 56℃

Sequence	Schedule	Remarks
First	< 6 EFPY	- Earlier one of [6EFPY, the time when the accumulated neutron fluence of the capsule exceeds 5X10^{22} n/m^2]
Second	9 EFPY	- 15 EFPY or at the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel inner wall location, whichever comes first - Earlier one of [15EFPY, 9.4 EFPY] = 32 EFPY/ 3.4 (lead factor)
Third	14 EFPY	- Not less than once or greater than twice the peak EOL vessel fluence - Between [1.0 EOL, 2.0 EOL] - If third surveillance is performed at 14 EFPY, 48 EFPY data would be obtained = 48 EFPY / 3.4 (lead factor)

Possible to obtain EOL(40 years) data

Possible to obtain CO(60 years) data

CO : Continued Operation (extended life to 60 yrs)

Withdrawal Schedule of KSNP Type Reactor

☐ Assumption
- 40 years design life
- predicted transition shift at vessel inside surface < 56℃

Sequence	Schedule	Remarks
First	< 6 EFPY	- Earlier one of [6EFPY, the time when the accumulated neutron fluence of the capsule exceeds $5X10^{22}$ n/m^2]
Second	15 EFPY	- 15 EFPY or at the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel inner wall location, whichever comes first - Earlier one of [15EFPY, 24.6 EFPY] = 32 EFPY/ 1.3 (lead factor)
Third	32 EFPY	- Not less than once or greater than twice the peak EOL vessel fluence - Between [1.0 EOL, 2.0 EOL] - If third surveillance is performed at 32 EFPY, 42 EFPY data would be obtained = 42 EFPY / 1.3 (lead factor)

Impossible to obtain EOL(40 years) data

Impossible to obtain CO(60 years) data

CO : Continued Operation (extended life to 60 yrs)

Obtaining EOL(40 years) Data

- Adjusted withdrawal schedule was submitted for Operation License Review

Sequence	Withdrawal Schedule (EFPY)											
	Hanbit				Hanul				Shin-Kori		Shin-Wolsong	
	3	4	5	6	3	4	5	6	1	2	1	2
First	6.20	6.02	6.49	6.54	6.61	6.57	6.93	7.16	6 *	6 *	6 *	6 *
Second	14.60	14.88	15*	15*	15.7	15*	15*	15*	15*	15*	15*	15*
Third	26*	26*	26*	26*	23*	23*	23*	23*	26*	26*	26*	26*
4th~6th	Standby	Standby	Standby	Standby	Standby	Standby	Standby	Standby	Standby	Standby	Standby	Standby

Source: FSAR of each plant. As of April 2018

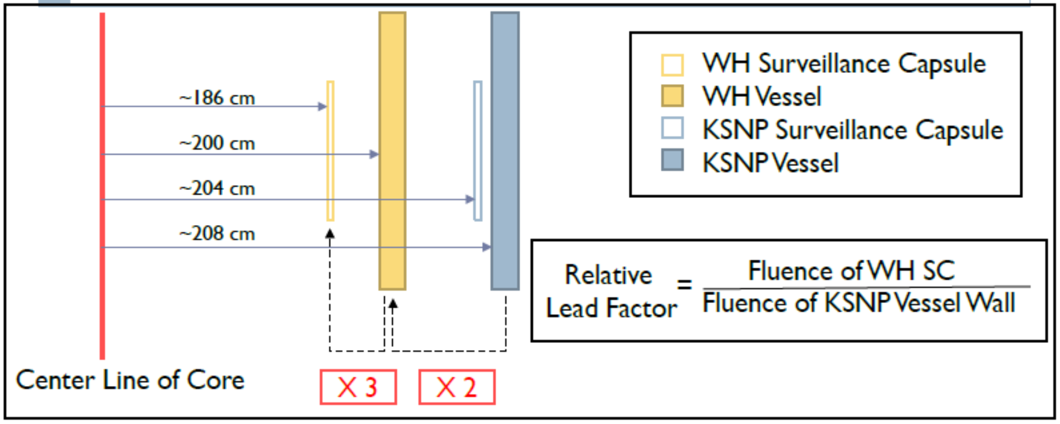
* : planned schedule (not performed)

Obtaining CO(60 years) Data (1)

- Accelerated Surveillance Test using WH reactors
  - New surveillance capsules were fabricated using archive materials of KSNP reactors, then inserted into WH reactors during 2014~2016

KSNP Reactors	WH Reactors	Remarks
Hanbit 3	Hanbit 1	two additional surveillance capsules per each KSNP reactor (one for 60 years, another for 80 years)
Hanbit 4		
Hanbit 5	Hanbit 2	
Hanbit 6		
Hanul 3	Kori 3	
Hanul 4		
Hanul 5	Kori 4	
Hanul 6		

Obtaining CO(60 years) Data (2)

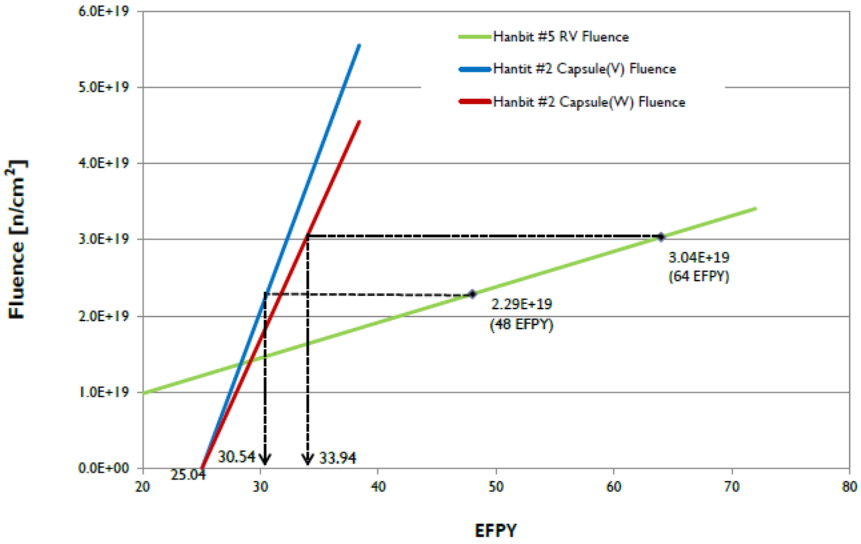


☐ WH Surveillance Capsule
▨ WH Vessel
☐ KSNP Surveillance Capsule
▨ KSNP Vessel

$$\text{Relative Lead Factor} = \frac{\text{Fluence of WH SC}}{\text{Fluence of KSNP Vessel Wall}}$$

Plant	Location		Flux	Relative Lead Factor*
Hanbit 2	Surveillance Capsule	V(107°)	1.32E+11	8.73
		W(110°)	1.08E+11	7.19
	RV Inner wall		3.59E+10	-
Hanbit 5	RV inner wall		1.50E+10	-

Obtaining CO(60 years) Data (3)



Issue 2: 60 years Design Life

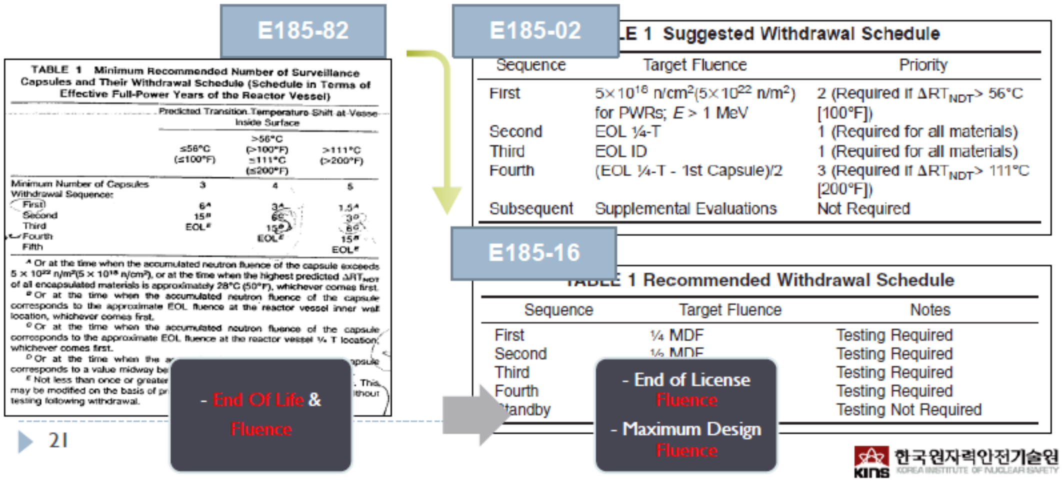
- Current withdrawal schedule
  - Based on 40 years design life
    - ASTM E185-82 7.6.2 "The withdrawal schedule is in terms of effective full-power years of the vessel with a design life of 32 EFPY"
  - Difficult to apply directly to 60 years design life reactors
  - Design life of APR 1400 reactor is 60 years

New criteria for withdrawal schedule is needed

- Design Life, Operating License Period, Continued Operation
  - Operating license period varies from country to country
    - (Korea) operating license period is determined by the design life of reactors
    - (USA) In AEA sec.103, "license shall be issued for a specified period,•••, but not exceeding 40 years
  - Design Life, LR(License Renewal), SLR(Subsequent License Renewal)…

Issue 2: 60 years Design Life

- Revision of Withdrawal Schedule in ASTM E185
  - Based on combination of Reactor Year and Fluence (E185-82)
  - Based on Fluence (since E185-02)



E185-82

TABLE 1 Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule (Schedule in Terms of Effective Full-Power Years of the Reactor Vessel)

	Predicted Transition Temperature Shift at Vessel Inside Surface		
	≤56°C (≤100°F)	>56°C (>100°F) ≤111°C (≤200°F)	>111°C (>200°F)
Minimum Number of Capsules	3	4	5
Withdrawal Sequence:			
First	6^A	3^A	1.5^A
Second	15^B	6^C	3^D
Third	EOL^E	15^B	6^C
Fourth		EOL^E	15^B
Fifth			EOL^E

[A] Or at the time when the accumulated neutron fluence of the capsule exceeds 5 × 10^{22} n/m^2(5 × 10^{18} n/cm^2), or at the time when the highest predicted ΔRT_{NDT} of all encapsulated materials is approximately 28°C (50°F), whichever comes first.

[B] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

[C] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel ¼ T location, whichever comes first.

[D] Or at the time when the a… capsule corresponds to a value midway be…

[E] Not less than once or greater… . This may be modified on the basis of pr… ithout testing following withdrawal.

- End Of Life & Fluence

E185-02

LE 1 Suggested Withdrawal Schedule

Sequence	Target Fluence	Priority
First	5×10^{18} n/cm^2(5×10^{22} n/m^2) for PWRs; *E* > 1 MeV	2 (Required if ΔRT_{NDT}> 56°C [100°F])
Second	EOL ¼-T	1 (Required for all materials)
Third	EOL ID	1 (Required for all materials)
Fourth	(EOL ¼-T - 1st Capsule)/2	3 (Required if ΔRT_{NDT}> 111°C [200°F])
Subsequent	Supplemental Evaluations	Not Required

E185-16

TABLE 1 Recommended Withdrawal Schedule

Sequence	Target Fluence	Notes
First	¼ MDF	Testing Required
Second	½ MDF	Testing Required
Third		Testing Required
Fourth		Testing Required
Standby		Testing Not Required

- End of License Fluence
- Maximum Design Fluence

Proposed Withdrawal Schedule in Korea

		Predicted Transition Temperature Shift at Vessel Inside Surface		
		≤ 56℃ (≤100 ℉)	>56 ℃(≤100 ℉) ≤ 111 ℃ (≤200 ℉)	〉 111℃ (>200 ℉)
Minimum Number of Capsules		4 ~~3~~	4	5
Withdrawal Sequence	First	A ~~6^A~~	A ~~3^A~~	A ~~1.5^A~~
	Second	C ~~15^B~~	C ~~6^C~~	D ~~3^D~~
	Third	B ~~EOL^E~~	B ~~15^B~~	C ~~6^C~~
	Fourth	E ~~-~~	E ~~EOL^E~~	B ~~15^B~~
	Fifth	-	-	E ~~EOL^E~~

A ~~Or~~ at the time when the accumulated neutron fluence of the capsule exceeds 5X10^{22} n/m^2 (5X10^{18} n/cm^2), or at the time when the highest predicted ΔRT_{NDT} of all encapsuled materials is approximately 28℃(50℉), whichever comes first.

B ~~Or~~ at the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel inner wall location~~, whichever comes first~~.

C ~~Or~~ at the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel ¼ T location~~, whichever comes first~~.

D ~~Or~~ at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

E Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.

Proposed Withdrawal Schedules in Korea

- Application to APR-1400 model

❑ Assume
- Lead Factor : 1.4(APR-1400)
- Calculation of Fluence at ¼ T location: RG-1.99 method $f = f_{surf}(e^{-0.24x})$

Sequence	Schedule	Remarks
First	< 6 EFPY	- At the time when the accumulated neutron fluence of the capsule exceeds 5X10^{22} n/m^2]
Second	17 EFPY	- At the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel ¼ T location $= 48 \times \frac{1}{\cong 2} \times \frac{1}{1.4}$ $f = f_{surf}(e^{-0.24x}) \rightarrow$ f @ ¼T= ~0.5f_{surf}
Third	34 EFPY	- At the time when the accumulated neutron fluence of the capsule corresponds to the approximately EOL fluence at the reactor vessel inner wall location $= 48 \times \frac{1}{1.4}$ (design life/lead factor)
4th~6th	Standby	

Concluding Remarks

- Surveillance test requirement in Korea is presented
  - current surveillance test is based on ASTM E185-82
- Low value of Lead Factor issue was resolved by
  - adjusting withdrawal schedule of KSNP type reactor (40 yrs data)
  - using WH reactor for acceleration surveillance test (60 yrs data)
- Revised withdrawal schedule is proposed to cover all the plant regardless of design life (in processing)

B.16 Ongoing Researches in Age-Related Degradation of Reactor Materials in Korea (B-S Lee)

USNRC LTO Metals Workshop 2019

www.kaeri.re.kr

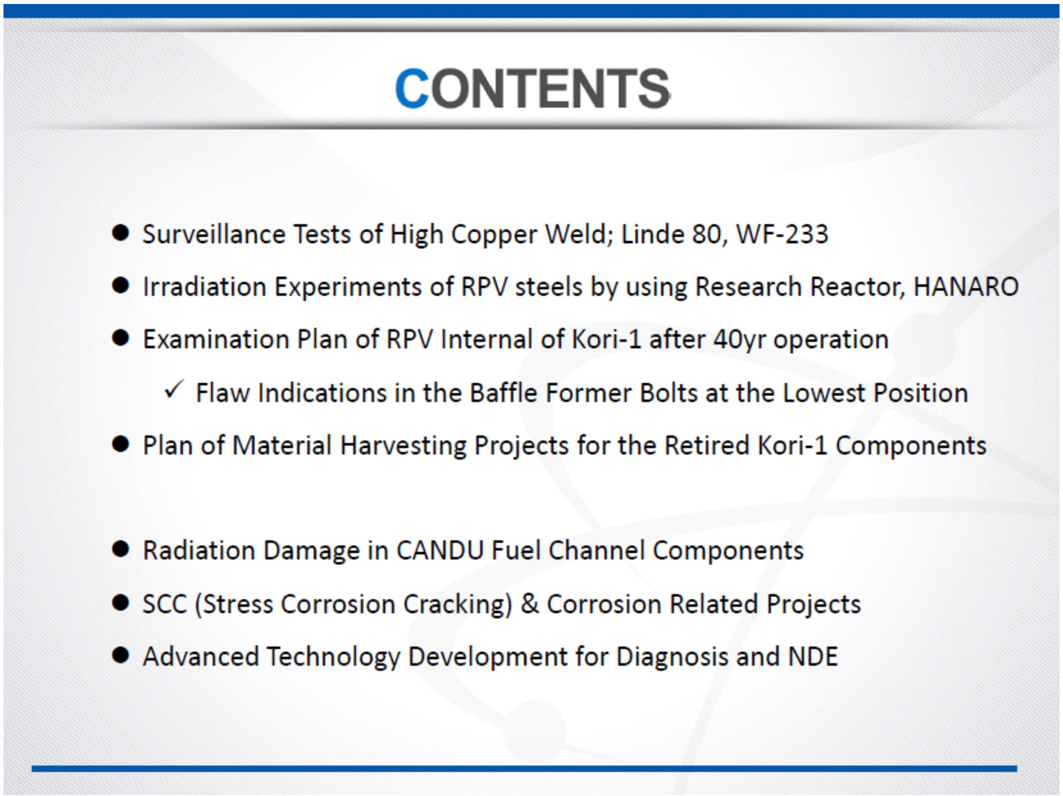
Ongoing Researches in Age-Related Degradation of Reactor Materials in Korea

<Int. Workshop on age-related degradation of reactor vessels and internals >

23~24, May 2019, USNRC Rockville, MD, USA

*Bong-Sang Lee

Korea Atomic Energy Research Institute



CONTENTS

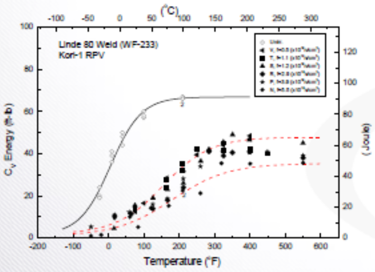
- Surveillance Tests of High Copper Weld; Linde 80, WF-233
- Irradiation Experiments of RPV steels by using Research Reactor, HANARO
- Examination Plan of RPV Internal of Kori-1 after 40yr operation
    - ✓ Flaw Indications in the Baffle Former Bolts at the Lowest Position
- Plan of Material Harvesting Projects for the Retired Kori-1 Components

- Radiation Damage in CANDU Fuel Channel Components
- SCC (Stress Corrosion Cracking) & Corrosion Related Projects
- Advanced Technology Development for Diagnosis and NDE

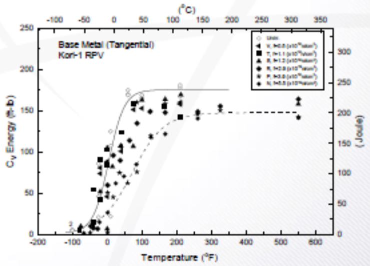
1 Surveillance Test Data of a High Copper Linde 80 Weld

- Kori-1 : 590MWe PWR, W/H 2-loop (1977~2017, 40 yr operation)
- Beltline circumferential weld: 0.23% Cu, Linde 80, WF-233
- The six surveillance capsules were all tested.
- Low USE (<50 ft-lb) & High ΔRT_{NDT} (RT_{PTS}>300°F)
- J-R tests by using modified 1X-WOL specimens
- M/C (T_o) tests by using reconstituted precracked Charpy specimens

Weld (Linde 80, WF-233)



Base Forging (SA508-Cl.2)



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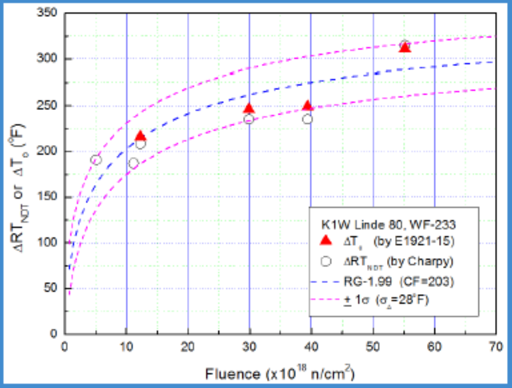
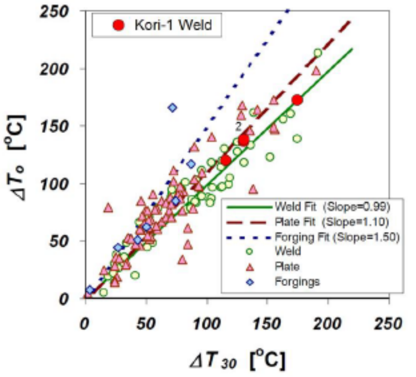
1 Transition temperature shifts by neutron irradiation



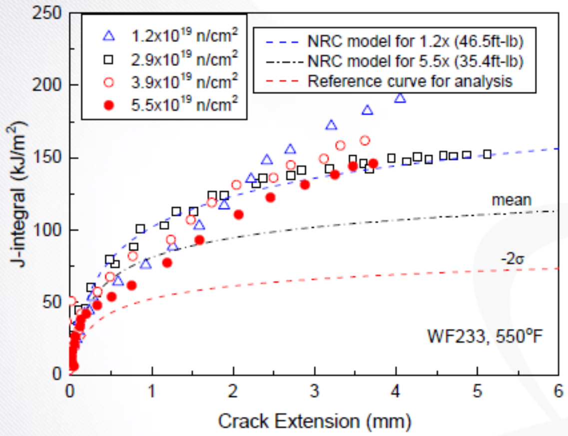
$\Delta T_{To} \approx \Delta T_{cv,30ft\text{-}lb}$ for the Linde 80 weld

Scatter was smaller in T_o values than Charpy.

by NUREG-1807

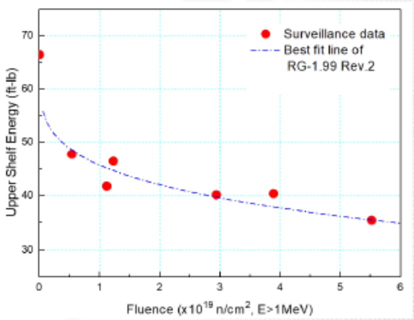


1 J-R curves from modified 1X-WOL specimens at 550°F



USNRC RG-1.161 database models could bound the K1W test data.

USE decreases according to RG1.99

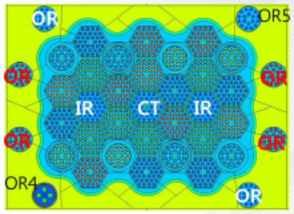


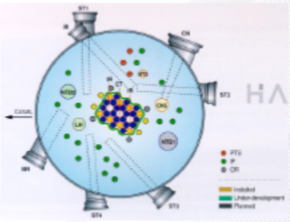
2 Irradiation Test Data of RPV steels using HANARO

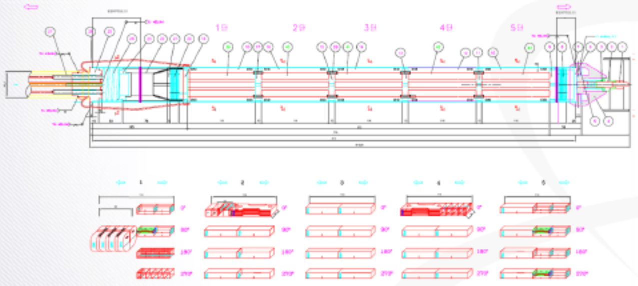
Instrumented capsule for material irradiation test

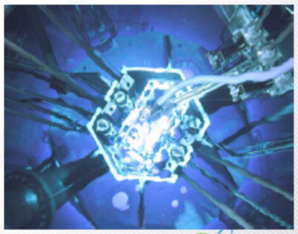
- ◆ CT hole flux: 0.7~1.5x10^{14} n/cm^{2} · sec
- ◆ OR hole flux: 0.6~1.7x10^{13} n/cm^{2} · sec

A capsule accommodates about 40 Cv specimens.









2 Summary of RPV Steels Irradiation Tests at HANARO

Capsule	Period	Fluence : $x10^{19}$ n/cm^2 (>1MeV)	Irr. Temp.: ℃	Materials	Irr. Hole	Remark
98M-02K	99.7.9~7.11	1.1~2.3	271~297	SA508 Gr.3	CT	RPV steel for OPR1000
99M-01K	00.6.14~6.17	1.3~2.9	272~305	SA508 Gr.3	IR2	
99M-02H	00.5.31~6.3	1.4~2.9	281~309	SA508 Gr.3	IR2	
00M-02K	01.5.2~5.6	1.2~3.3	282~309	SA508 Gr.3	IR2	
02M-02K	03.8.15~8.21	3.0~6.4	271~297	SA508 Gr.3	CT	
08M-01K	08.10.8~10.15	3.9~8.3	255~317	SA508 Gr.4N	CT	Model Alloy
08M-02K	09.4.7~4.28, 4.30~5.3	1.3~4.4	269~310	SA508 Gr.4N	OR5	
11M-25K	12.2.24~3.7	7.3~15.3	265~320	SA508 Gr.4N	CT	Model Alloy
13M-02K	13.5.27~6.24, 7.1~7.28	2.4~8.9	275~301	SA508 Gr.3	OR5	RPV steel for OPR1000 (High fluence effects)
16M-01K*	18.5.25~6.24, 7.9~7.30	$1.9\sim7.1x10^{19}$	290	SA508 Gr.3	OR4	RPV steel for APR1400
17M-01K*	18.11.20~11.28	$4.6\sim9.1x10^{19}$	194~249	SA533 B1	CT	RPV steel for WH RPV in Korea

Specimens : Charpy, PCVN, Tensile, CT, mini-PCVN, Small Punch, etc.

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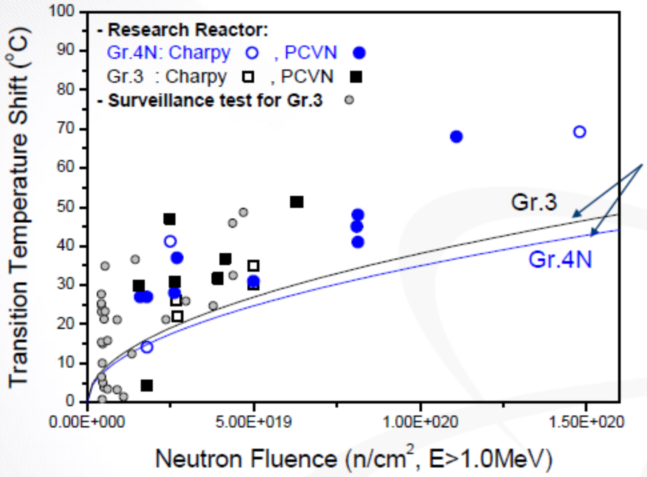
2 Irradiation Embrittlement Trend of Korean Forgings

◆ **SA508 Gr.3 steels & Gr.4N model alloys**



Trend curves
(EONY model)

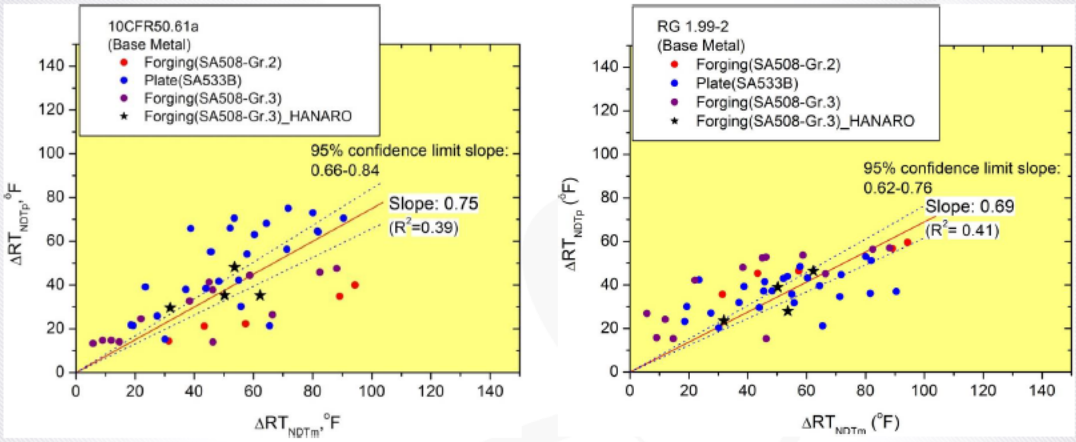
ΔT_{41J} = MD + CRP

- Low Cu content
- Only MD was considered

⇒ Overall irradiation embrittlement trend of SA508 Gr.4N model alloys was similar to that of SA508 Gr.3 steels
Similar result was reported previously (*G. Wire et al., ASTM STP 1447, 2004)

⇒ Ni content in SA508 Gr.4N is higher, but P & Mn contents are low

2 Embrittlement Prediction of Surveillance Data

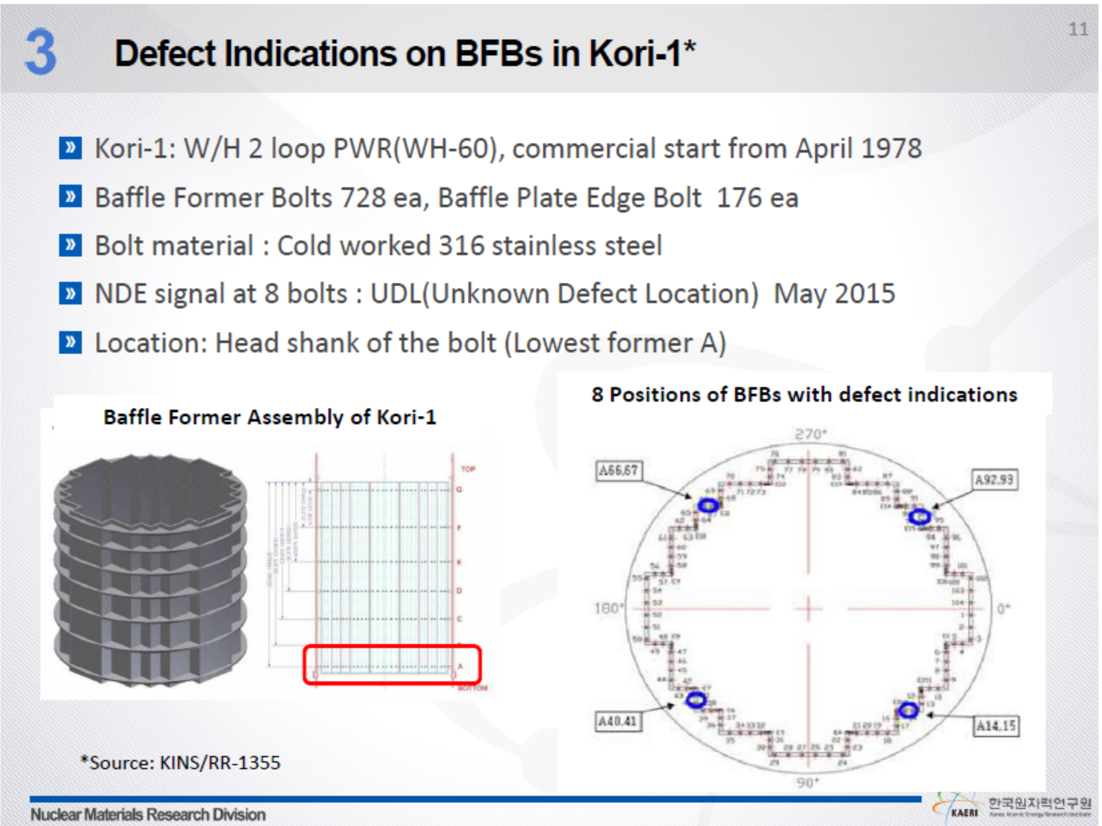


- Measured shifts are bigger than the model predictions
- HANARO research reactor data are comparable to plant data.

(even though the flux range is about three orders higher, ~10^{14}n/cm^2.sec)

3 Examination Plan of RPV Internal of Kori-1 after 40 yr

- Defect indications on 8 baffle-former-bolts at Kori-1 RV internal (May 2015)
  - Defect signals were found at a lower dose level position.
- Kori-1 was permanently shutdown in 2017 after 40 yr operation.
- The bolts of defect signals will be removed for destructive analysis.
  - To identify the actual defect morphology of the signal indications
  - For better understanding on degradation(IASCC) mechanism of BFB materials
  - For better modelling of IASCC parameters, such as dose and stress, etc.
  - To improve the aging management of RV internals
- Will be extended to the materials harvesting project for the retired Kori-1 components for evaluation of irradiation effect and materials degradation on the actual components(RPV, Internals, Nozzles, SG)



3

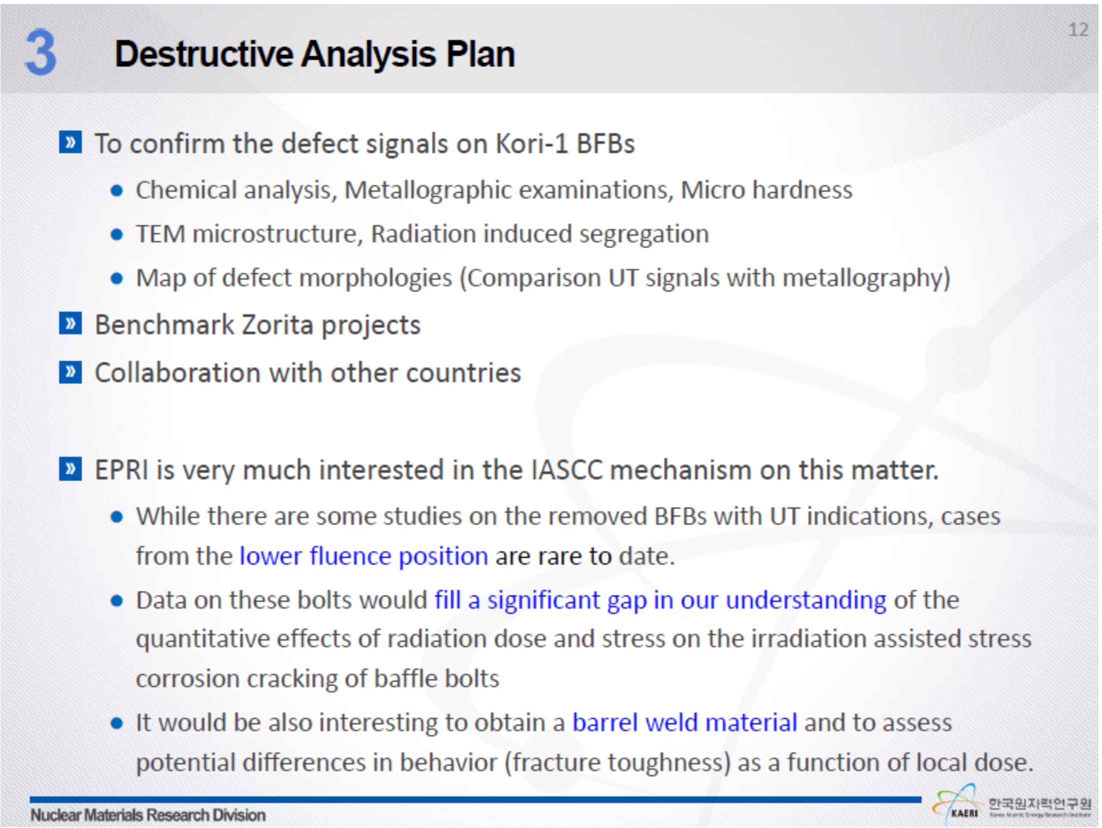
Defect Indications on BFBs in Kori-1*

- Kori-1: W/H 2 loop PWR(WH-60), commercial start from April 1978
- Baffle Former Bolts 728 ea, Baffle Plate Edge Bolt 176 ea
- Bolt material : Cold worked 316 stainless steel
- NDE signal at 8 bolts : UDL(Unknown Defect Location) May 2015
- Location: Head shank of the bolt (Lowest former A)

Baffle Former Assembly of Kori-1

8 Positions of BFBs with defect indications

*Source: KINS/RR-1355



3

Destructive Analysis Plan

- To confirm the defect signals on Kori-1 BFBs
  - Chemical analysis, Metallographic examinations, Micro hardness
  - TEM microstructure, Radiation induced segregation
  - Map of defect morphologies (Comparison UT signals with metallography)
- Benchmark Zorita projects
- Collaboration with other countries

- EPRI is very much interested in the IASCC mechanism on this matter.
  - While there are some studies on the removed BFBs with UT indications, cases from the lower fluence position are rare to date.
  - Data on these bolts would fill a significant gap in our understanding of the quantitative effects of radiation dose and stress on the irradiation assisted stress corrosion cracking of baffle bolts
  - It would be also interesting to obtain a barrel weld material and to assess potential differences in behavior (fracture toughness) as a function of local dose.

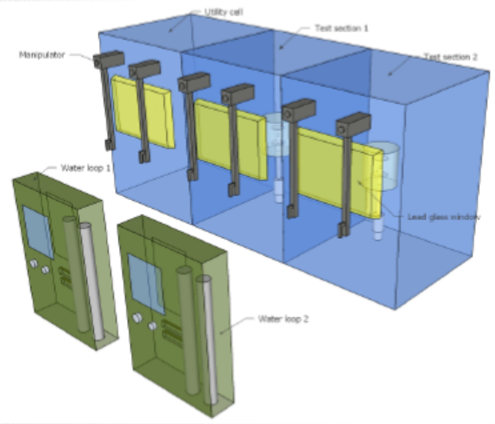
4 Plan of National Project on the Kori-1 Harvested Materials

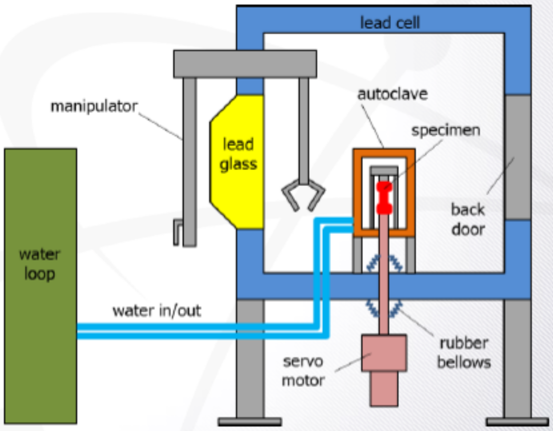
- Project title: 'Technology development for IASCC on reactor internals' under the program of 'Nuclear core technology development'
- Sep. 2019 through Aug. 2024, hopefully (Stage-I)
- Participation: KAERI, KHNP-CRI and other companies
- Deliverables
  - Specimen machining technology for highly radioactive materials
  - Test facility demonstration for IASCC experiments
  - Identification and characterization of the defect signals on Kori-1 BFBs
- Stage-II will include the other topics (after 2022)
  - Embrittlement of RPV thick wall & actual welds
  - SG tube integrity & SCC resistance of Alloy 690
  - Demonstration of the NDT capability : SG tube, Pipe Welds

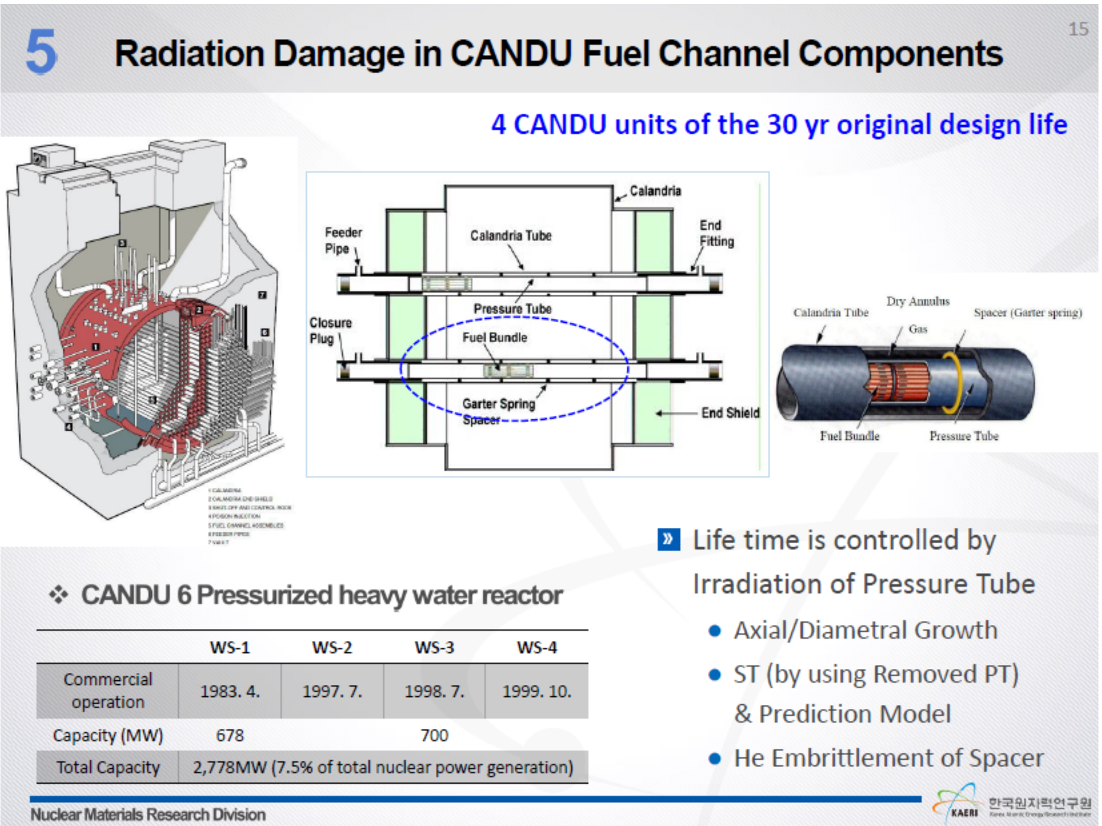
---

4 Facility Set Up for IASCC Experiments

- Lead-shielded hot cells are going to be constructed for IASCC tests.
- Design of the building and the hot cell facilities are going on.
- International cooperation on high dose materials testing experiences is necessary.







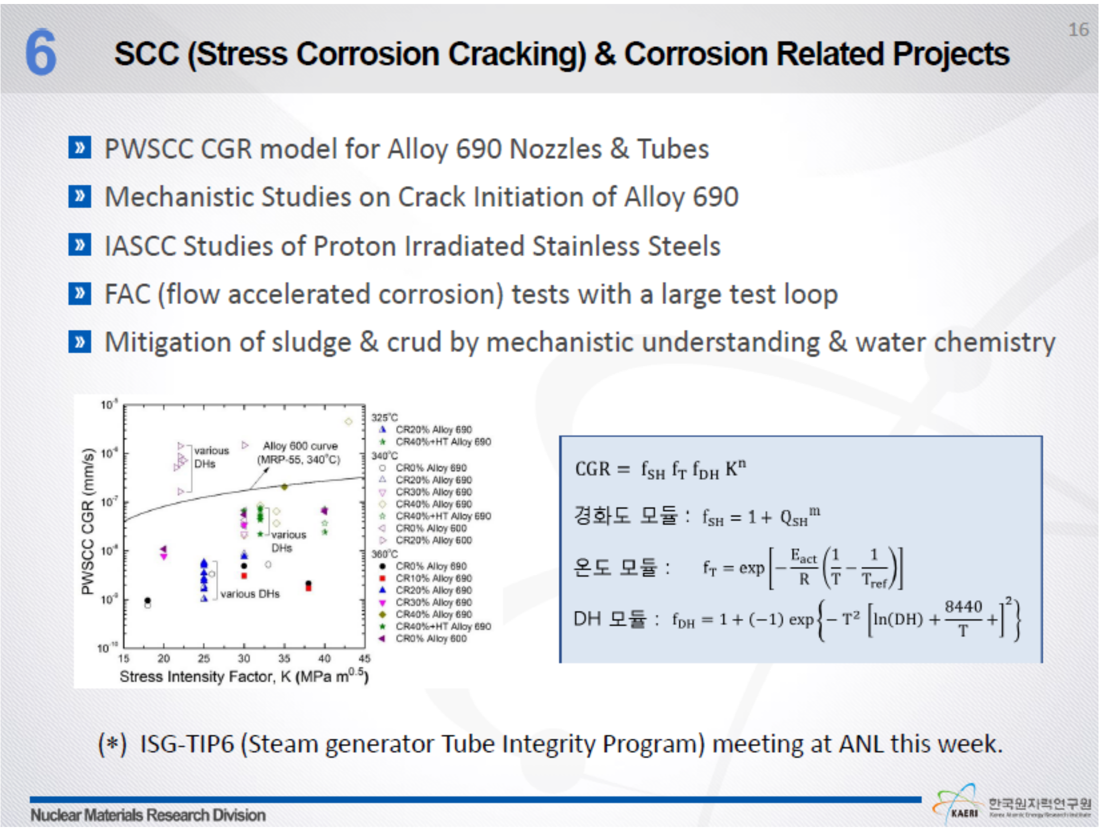
5 Radiation Damage in CANDU Fuel Channel Components

4 CANDU units of the 30 yr original design life

❖ CANDU 6 Pressurized heavy water reactor

	WS-1	WS-2	WS-3	WS-4
Commercial operation	1983. 4.	1997. 7.	1998. 7.	1999. 10.
Capacity (MW)	678	700		
Total Capacity	2,778MW (7.5% of total nuclear power generation)			

- Life time is controlled by Irradiation of Pressure Tube
  - Axial/Diametral Growth
  - ST (by using Removed PT) & Prediction Model
  - He Embrittlement of Spacer



6 SCC (Stress Corrosion Cracking) & Corrosion Related Projects

- PWSCC CGR model for Alloy 690 Nozzles & Tubes
- Mechanistic Studies on Crack Initiation of Alloy 690
- IASCC Studies of Proton Irradiated Stainless Steels
- FAC (flow accelerated corrosion) tests with a large test loop
- Mitigation of sludge & crud by mechanistic understanding & water chemistry

$$\mathrm{CGR} = f_{SH}\, f_T\, f_{DH}\, K^n$$

경화도 모듈 : $f_{SH} = 1 + Q_{SH}{}^m$

온도 모듈 : $f_T = \exp\left[-\frac{E_{act}}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right]$

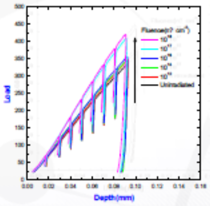
DH 모듈 : $f_{DH} = 1 + (-1)\exp\left\{-T^2\left[\ln(DH) + \frac{8440}{T} +\right]^2\right\}$

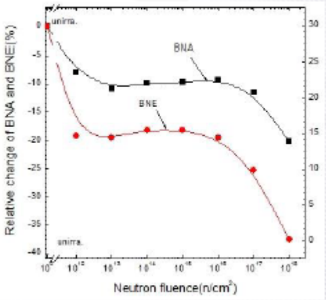
(*) ISG-TIP6 (Steam generator Tube Integrity Program) meeting at ANL this week.

7 Advanced Technology for Diagnosis and NDE

- Advanced ECT probe for diagnosis of steam generator U-bend
    - Evaluation of the effects of sludge & foreign object noise
- Non-linear UT techniques for cracks in pipe welds
- Long range guided wave technique for buried pipe inspection
- PEC (Pulsed Eddy Current) with Hall sensor array for diagnosis of pipe under insulation

- Basic Studies on Non-Destructive Characterization of Irradiation Embrittlement of RPV steels
    - Magnetic Barkhausen Noise
    - Maybe a possible tool
    - Needs samples of a series of actual neutron exposure





Data from TRIGA at 70ºC

SUMMARY

- Irradiation embrittlement data from surveillance tests and research reactor tests were summarized for Korean RPV steels.
- Some baffle-former-bolts of defect signals will be removed from Kori-1 internal and investigated in detail soon.
- Materials harvesting projects for the retired Kori-1 components are being prepared under the national nuclear safety research programs.
- Other important researches are also going on;
    - Radiation Damage in CANDU Fuel Channel Components
    - SCC (Stress Corrosion Cracking) & Corrosion Related Projects
    - Advanced Technology for Diagnosis and NDE

B.17 Operating Experience (OpE) on RV Internals, RV Head Penetrations, and RCS Small Bore Nozzles in Korea (J-S Yang)

Spring 2019 Workshop on Age-Related Degradation of Reactor Vessels and Internals
U.S. Nuclear Regulatory Commission Headquarters, May 23-24, 2019

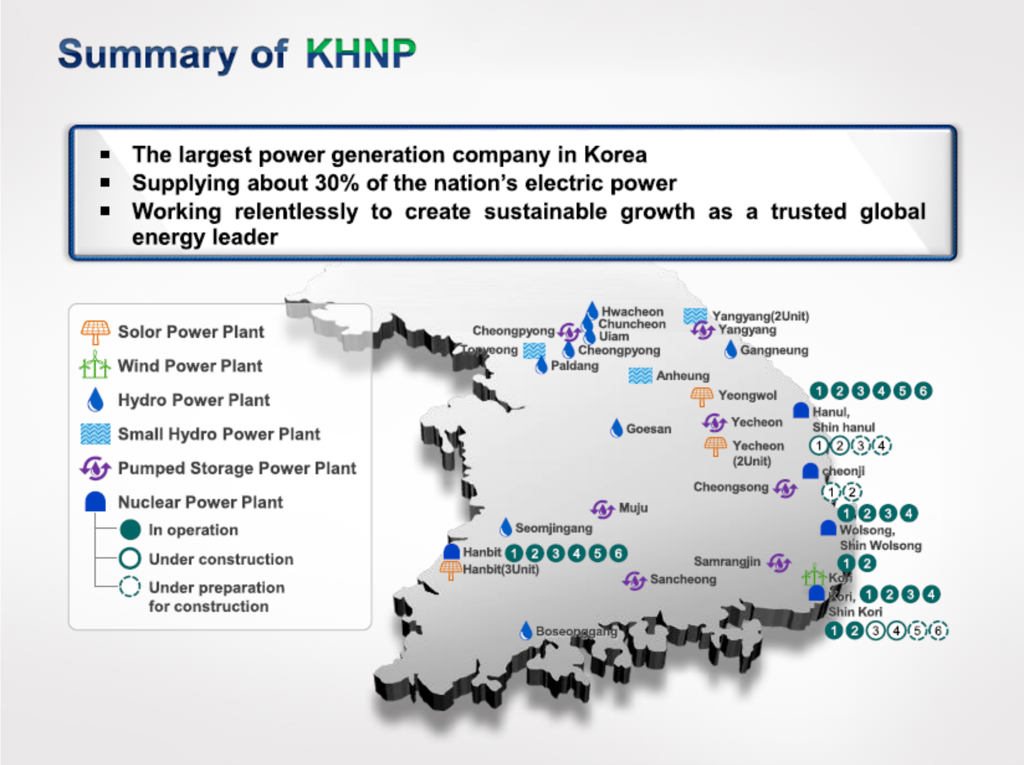
Operating Experience (OpE) on
RV Internals, RV Head Penetrations, and RCS Small Bore
Nozzles in Korea

May 24th, 2019

Jun-Seog Yang

* KHNP Central Research Institute

KHNP



Summary of KHNP

- The largest power generation company in Korea
- Supplying about 30% of the nation's electric power
- Working relentlessly to create sustainable growth as a trusted global energy leader

Contents

KHNP

TOPIC 1 OpE on Reactor Internal in KOREA

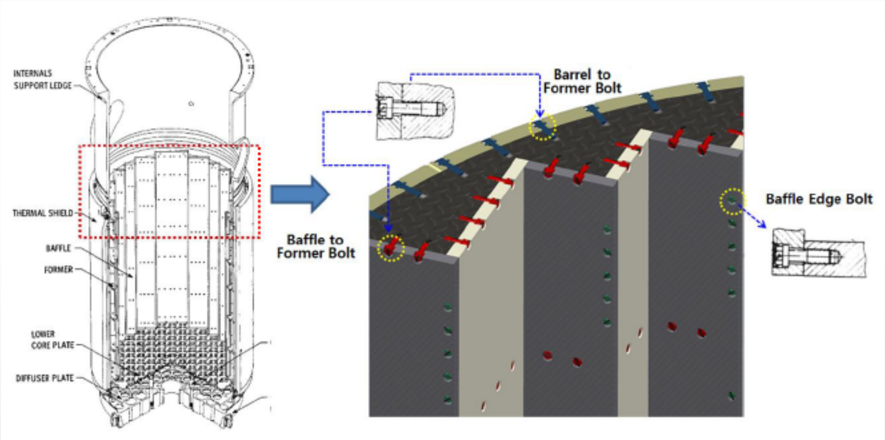
KHNP

2015 Kori Unit 1 Baffle Bolts Inspections

- Kori Unit 1, 2 loop Westinghouse reactor
  - Commercial operation in 1976 and had 40 years.
  - Shutdown permanently Since June 16th, 2017
- Bolts are 316 stainless steel with hex head design with a locking bar
- All 728 baffle to former blots were UT examined and VT-3 examined
- No back-wall signal from UT in eight baffle former bolts

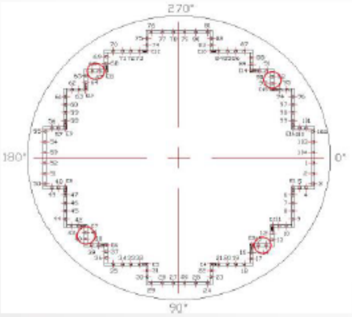
2015 Kori Unit 1 Baffle Bolts Inspections

- 5/18 ~25, 2015 : Inspection of UT for Baffle Bolts
  - Baffle Former Bolts (BFBs) : 728 EA (UT and VT)
  - Baffle Edge Bolts : 176EA (VT)



2015 Kori Unit 1 Inspection Findings on BFBs

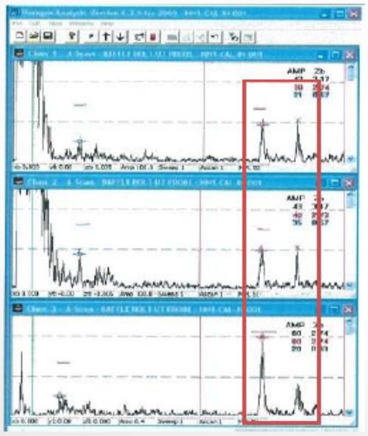
- 5/25, 2015 : No back-wall signals in 8 Baffle Former Bolts
- Defect unknown locations (DUL): 45°, 135°, 225°, 315° each 2 EA



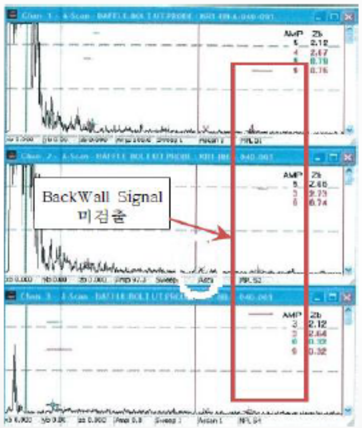
2015 Kori Unit 1 Inspection Findings on BFBs

- Analysis of Signals from UT(2 elements, 3channels)

Sound Bolts

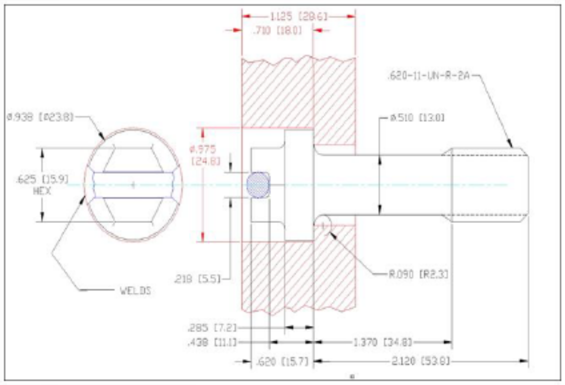


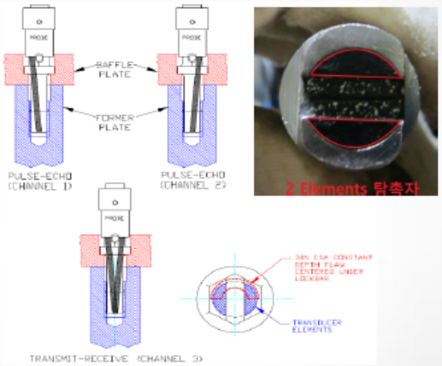
Defect Unknown Location



2015 Kori Unit 1 Inspection Findings on BFBs

- Signals from UT(2 elements, 3channels)

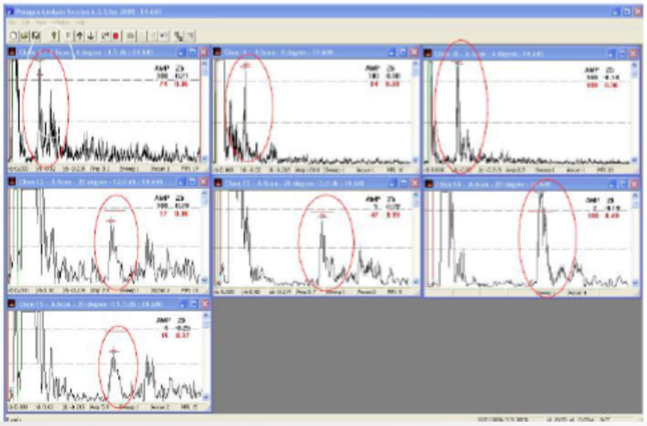




2015 Inspection Findings on BFBs of Kori Unit 1

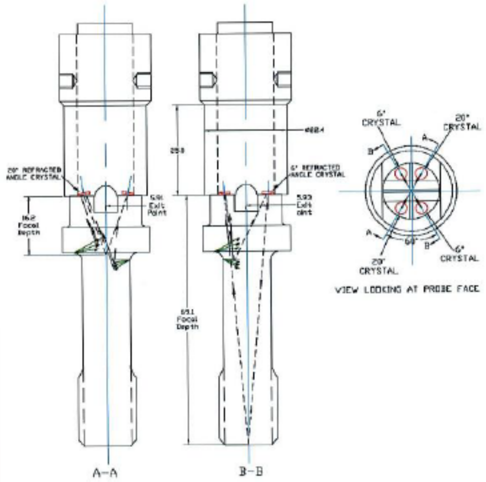
- Analysis of Signals from UT(4 elements, 16channels)

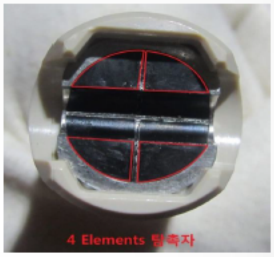
Defect Locations for bolts with DUL



2015 Inspection Findings on BFBs of Kori Unit 1

- Analysis of Signals from UT(4 elements, 16channels)
    - 20° Refracted Angle Crystal, 6° Refracted Angle Crystal





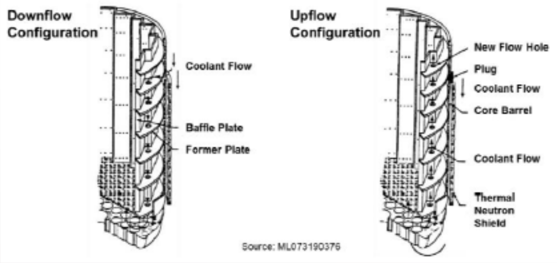
2015 Kori Unit 1 Inspection Findings on BFBs

- No back-wall signals in 8 BFBs were identified as DUL.
- UT with 4 Element probes found defects in shank which was located back locking bar.

Identification of Bolts	Inspection Results for Baffle Former Bolts		
	1999	2006	2015
	FRAMATOME	KPS	KPS
A14	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A15	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A40	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A41	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A66	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A67	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A92	Back Wall Signal	Back Wall Signal	Defect Unknown Location
A93	Back Wall Signal	Back Wall Signal	Defect Unknown Location

2015 Kori #1 Inspection Findings on BFBs

- OpE of Coolant Flow : Converted Up-flow
  - WEC plants have a "down-flow" baffle barrel region design, fuel degradation due to baffle jetting could be an indicator of potential baffle bolt degradation.
  - WEC plants which were the most susceptible to baffle-jetting related fuel degradation have been converted to up-flow.
  - Kori Uint 1 was converted to up-flow in 4 cycle operation after commercial operation in 1976.

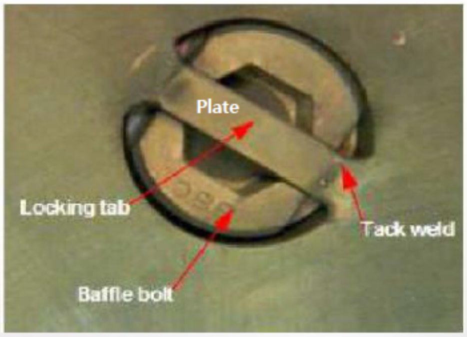


Source: ML073190376

2015, Kori #1 Inspection Findings on BFBs

- Comparison of Bolt Design
  - Westinghouse 4 Loop, down flow reported lots of bolt degradation.
  - D.C. Cook 2, 4 Loop, down-flow : Plate type locking bar, tack weld to baffle plate.
  - Kori 1, 2 Loop, converted up-flow: Round type locking bar, welded to baffle plate.

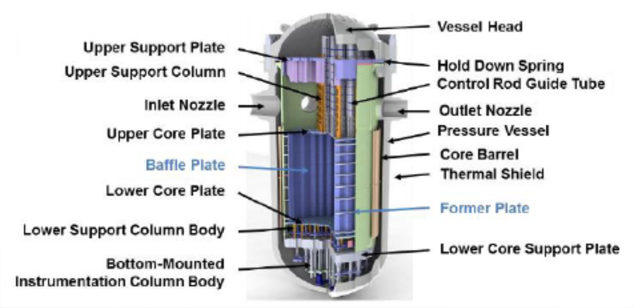
Kori #1



D.C. Cook #2

Reactor Internals Inspection Program Overview

- To Develop guidance to manage aging on RV internals in Korea
- MRP-227 I&E Guideline : What and When is to be inspected
  - Inspection and evaluation requirements for reactor internal components
- MRP-228 Inspection Standard : How to inspect
  - NDE methodology requirements to support MRP-227



Post 2015 Kori #1 Inspection Findings on BFBs

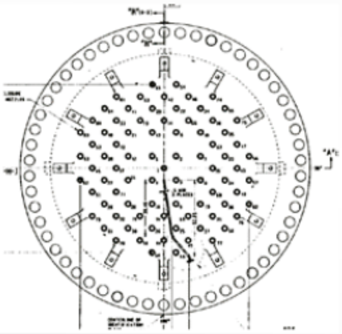
- 2016 : 100% VT and 25% UT Inspection for 200 bolts were done with no additional indications.
- 2020 : Pull 16 bolts out from Kori#1
- 2021 : Inspection and evaluation procedures of reactor internals
- 2023 : Hot cell testing of BFBs in KAERI

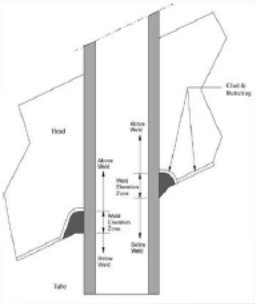
TOPIC 2 OpE on RV Head Penetrations in KOREA

KHNP

2012 RVHP Inspection Results in Hanbit#3

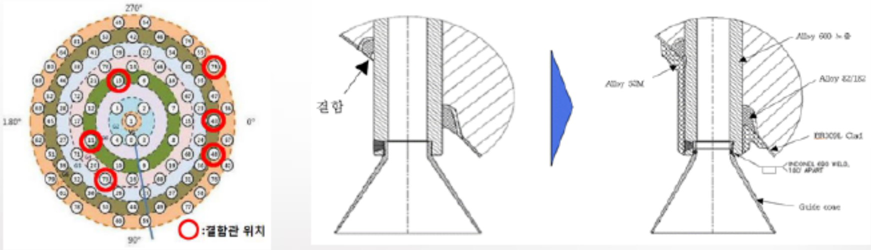
- Hanbit#3 started the commercial operation in 1995
- 11/12, 2012 : Inspection of UT for RVHP
  - Reactor Head Penetrations : 85EA (UT and ET)





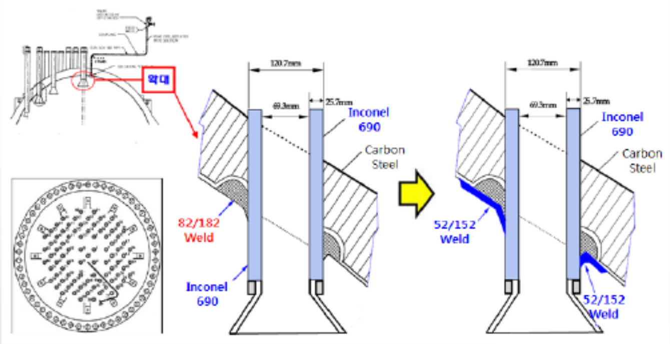
2012 RVHP Inspection Results of Hanbit#3

- UT detected six flaws : P11, P43, P70, P75, P13, P48
- All flaws were oriented axially and located in CRDM tube near the location of the J-groove weld toe
- Liquid penetrant examinations confirmed the existence, location, and orientation of the identified flaws and none of the flaws presented a leakage path to the upper surface of the RVHP
- Six penetrations were repaired with the embedded flaw technique



POST 2012 RVHP Inspection Results of Hanbit#3

- Replacement of Reactor Vessel Closure Head
  - CRDM Nozzles(Alloy 600→690), J-groove weld(82/182→52/152)
  - Hanbit#3&4(2015), Hanwl#1&2(2022)
- Preventive Maintenance of Reactor Vessel Closure Heads
  - CRDM Nozzles(Alloy 690), J-groove weld(82/182→52/152)
  - Hanwul#3&4,Hanbit#5&6(~2022) using seal weld technique



TOPIC 3 OpE on RCS Small Bore Nozzles in KOREA


KHNP

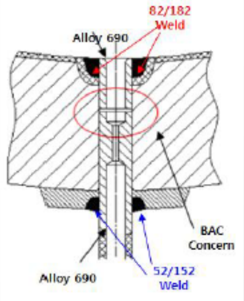
2016 Hanwul#3 Small Diameter Nozzle Inspection

- Hanwul Unit 3, 2 loop CE reactor
  - Commercial operation in 1996 and had ~20 EFPY
- VT Inspection of instrument nozzles in RCS Hot Leg
  - Temperature nozzle : 10EA, Sampling nozzle : 1E
  - Pressure measure nozzle : 8EA
- Nozzle Material : Tube(Alloy 690), Weld(82/182)
- A Leak was discovered from sampling nozzle in Mar. 2016

2016 Hanwul#3 Inspection Results of Sampling Nozzle

- Leak was discovered from sampling nozzle during visual inspection for evidence of boric acid deposits
  - Crack was short, axial and located near the J-groove welds to be attached nozzle to the hot leg piping

- The leak was repaired using half nozzle repair
- The plant qualified a UT technique to inspect the low alloy steel base metal for general corrosion
- There has been no indication of general corrosion of low alloy steel base metal



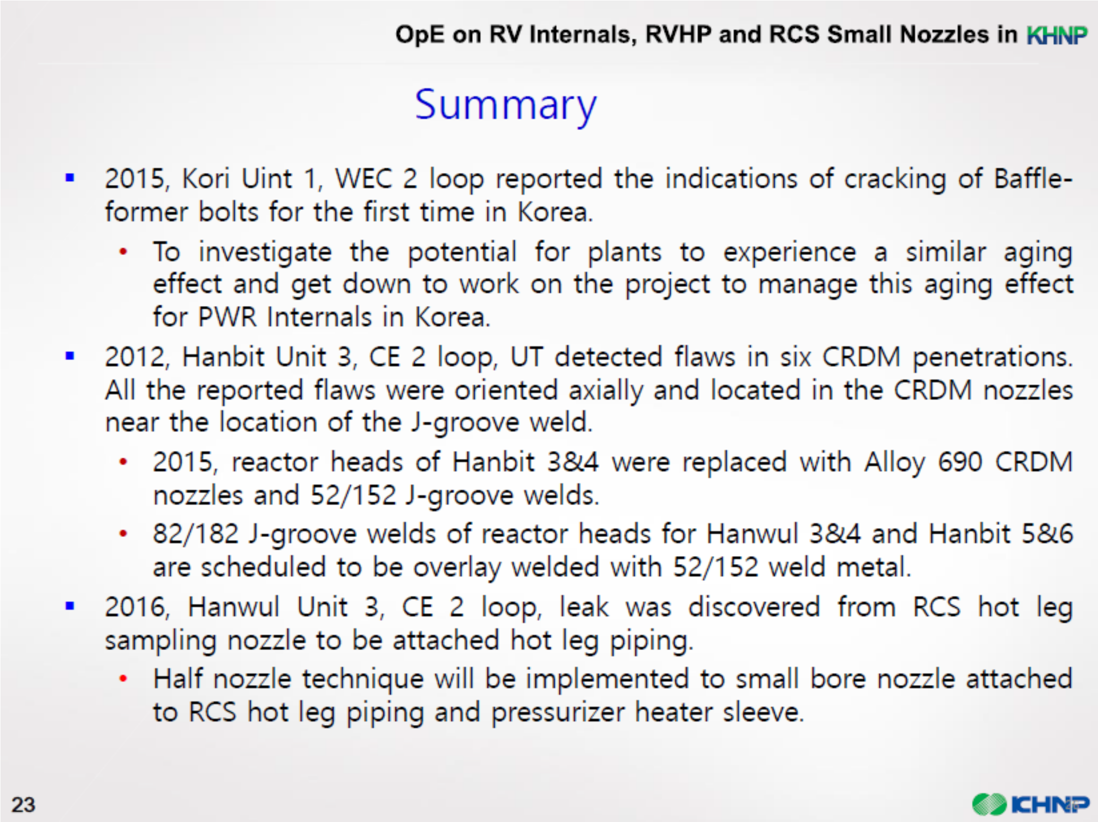
POST 2016 Hanwul#3 Sampling Nozzle Leak

- 1st 2019 ~ 2021 : Small bore nozzles of RCS Hot Leg Pipings
  - Nozzles(Alloy 690), J-groove weld(**82/182→52/152**)
  - Hanbit#3&4, Hanwul#3&4, Hanbit#5&6 (**total 113EA**)
  - Preventive Maintenance : Half nozzle repair with pad or nod-pad
- 2nd 2021 ~ 2024 : Heater sleeve nozzles of Pressurizer
  - Nozzles(Alloy 690), J-groove weld weld(**82/182→52/152**)
  - Hanbit#3&4, Hanwul#3&4, Hanbit#5&6 (**total 258EA**)
  - Preventive Maintenance : Half nozzle repair with pad or nod-pad

Chapter 3

Conclusions

KHNP



Summary

- 2015, Kori Uint 1, WEC 2 loop reported the indications of cracking of Baffle-former bolts for the first time in Korea.
  - To investigate the potential for plants to experience a similar aging effect and get down to work on the project to manage this aging effect for PWR Internals in Korea.
- 2012, Hanbit Unit 3, CE 2 loop, UT detected flaws in six CRDM penetrations. All the reported flaws were oriented axially and located in the CRDM nozzles near the location of the J-groove weld.
  - 2015, reactor heads of Hanbit 3&4 were replaced with Alloy 690 CRDM nozzles and 52/152 J-groove welds.
  - 82/182 J-groove welds of reactor heads for Hanwul 3&4 and Hanbit 5&6 are scheduled to be overlay welded with 52/152 weld metal.
- 2016, Hanwul Unit 3, CE 2 loop, leak was discovered from RCS hot leg sampling nozzle to be attached hot leg piping.
  - Half nozzle technique will be implemented to small bore nozzle attached to RCS hot leg piping and pressurizer heater sleeve.

THANK YOU

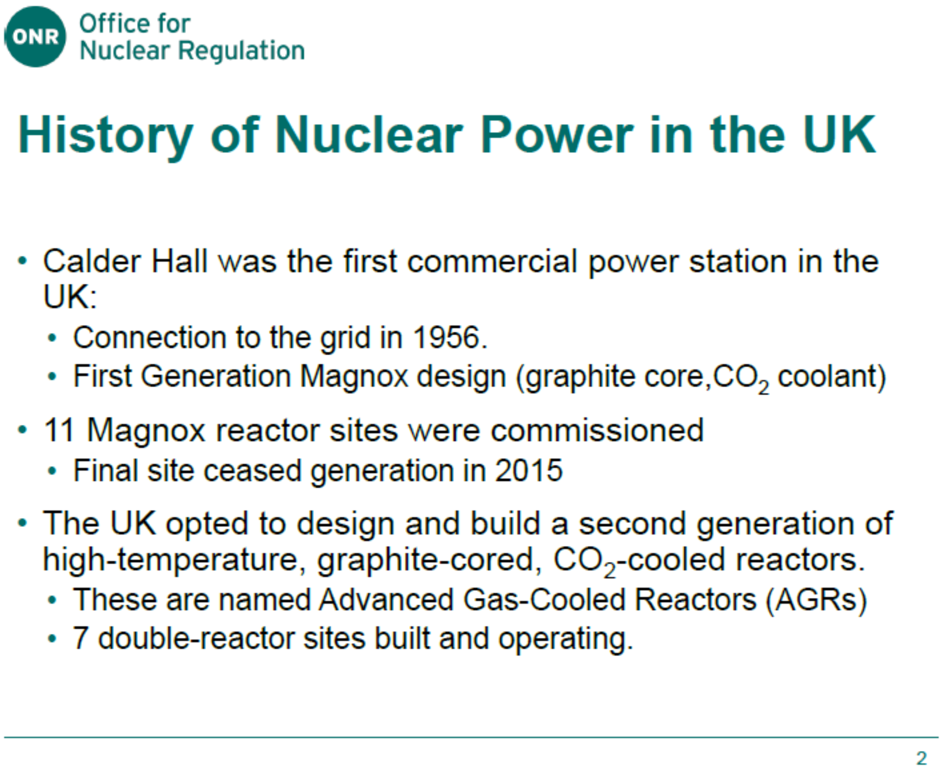
B.18 UK Regulatory Experience in Materials Ageing (G. Hopkin)

Office for
Nuclear Regulation

UK Regulatory Experience in Materials Ageing

Gareth Hopkin

Office for Nuclear Regulation (UK)



Office for
Nuclear Regulation

History of Nuclear Power in the UK

- Calder Hall was the first commercial power station in the UK:
  - Connection to the grid in 1956.
  - First Generation Magnox design (graphite core,CO_2 coolant)
- 11 Magnox reactor sites were commissioned
  - Final site ceased generation in 2015
- The UK opted to design and build a second generation of high-temperature, graphite-cored, CO_2-cooled reactors.
  - These are named Advanced Gas-Cooled Reactors (AGRs)
  - 7 double-reactor sites built and operating.

Civil Light Water Reactors in the UK

- In the 1980s, a government strategic decision was made to pursue light-water technologies.
- The Sizewell B PWR reactor was the first of an intended fleet of PWR reactors, but was the only one constructed.
  - Westinghouse 4-loop design
  - Build of the nuclear island led by Framatome
  - Build Started in 1987
  - Connection to the grid 1995

- The Marshall Light-Water Study Group, led by Walter Marshall, was influential in setting UK expectations for Structural Integrity of these plants.

Operating Plant – Irradiation Embrittlement

- The Marshall Light-Water Reactor Study Group contained advice for the management of irradiation embrittlement:.
  - A surveillance programme was initiated at SZB, including the following items:
    - Charpy Samples
    - Fracture toughness test coupons
    - Samples of pre-strained materials

  - The programme was for the original life of the station (40y)
  - On extension of the plant life, the programme has been extended.

Civil Light Water Reactors in the UK

- The UK has a process of Generic Design Assessment (GDA) for new nuclear build in the UK
  - Getting a Design Acceptance Certificate (DAC) does not mean that the design is licensed for build.
  - The UK design expectations within Structural Integrity include expectations for monitoring materials properties through-life.
  - The following designs have achieved a DAC:
    - EDF Areva EPR
    - Westinghouse AP1000® nuclear power plant
    - Hitachi-GE UK-ABWR
  - The HPR-1000 by General Nuclear Systems is at Step 3 of GDA
- ONR is performing initial assessments on Small Modular Reactors and Advanced Nuclear Technologies for the UK government.

Safety Assessment Principles (SAPs)

- The ONR Safety Assessment Principles (SAPs) are:
  - High level expectations for plant built or operating in the UK.
  - Goal-setting regime means that these expectations can be met by any route, at the discretion of the duty-holder.
  - EAD.03 relates directly to materials ageing, including irradiation embrittlement:

Engineering principles: ageing and degradation	Periodic measurement of material properties	EAD.3
Where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties.		

  - The properties should be obtained from fully representative samples of the material especially when the component or structure performs a principal role in ensuring nuclear safety.

Safety Assessment Principles (SAPs)

- In practice EAD.03 sets the expectation that there will be a suitable and sufficient programmes in place to monitor:
  - Thermal Ageing
  - Irradiation Embrittlement
- The scope and depth of the programmes should be proportionate to the nuclear safety of the system, structure or component.
- There is, for components of the highest reliability, that the surveillance programmes should be fully representative of the plant materials and environment.

Operating Plant – Irradiation Embrittlement

- The Safety Case for the plant is based around a demonstration of defect tolerance
- Safety Case for SZB is based upon the results of the fracture toughness specimens.
  - Charpy results confirm ASME code compliance, but are not central to the Safety Case for operation for the plant.
- Pre-strained data inform the understanding, but do not contribute directly to the demonstration of defect tolerance.
- The licensee aspire to extend the life of the plant beyond the original 40y lifespan.

Operating Plant – Irradiation Embrittlement

- Pre-strained samples
  - Included at the suggestion of the Light-Water Reactor Study Group.
  - Pre-strain intended to represent localised strains from pressure testing and manufacture.
  - Pre-strained was present for both Charpy and Fracture Toughness samples

- Pre-Strained samples bring down the fracture toughness and Charpy results, but quantification of the effect is difficult.

New Build Plant

- The Generic Design Assessment (GDA) process does not licence plant for build in the UK, it provides requesting parties with confidence that the plant has no fundamental issues to prevent UK build.
  - ONR does not licence technologies
  - ONR does permission activities

- The management of materials ageing must be addressed as part of the GDA process.
  - The detail necessary in the submission is directly dependent upon the nuclear safety of the component.
  - The UK interpretation of ALARP is applicable.

New Build Plant – UK ABWR

- ONR placed the following Action upon Hitachi-GE:
  - Regarding Materials compositions, Hitachi-GE should provide evidence that:
    - Relevant Good Practice regarding Materials compositions has been considered fully, notably from previous GDAs.
    - Irradiation Embrittlement surveillance programmes are adequate to mitigate the risk of irradiation embrittlement through-life.

- The requesting party proposed an irradiation embrittlement surveillance programme that was:
  - In excess of that required by code
  - As large as possible, without requiring redesign of core components.
  - Backed up by modelling to international codes

New Build Plant – UK EPR™

- The design of the UK EPR™ includes a heavy reflector to prevent excess embrittlement of the RPV steel.
- This has been the subject of a GDA issue:
  - Demonstration that the principles of the surveillance scheme adequately take account of the implications of the differences in neutron energy spectra between the location of the specimens and the RPV wall. This is expected to include the following activities:
    - Provision of evidence showing that the principles of the surveillance scheme adequately take account of the implications of the differences in neutron energy spectra between the location of the specimens and the RPV wall;
    - Justification of the concepts inherent in the analysis and interpretation of the surveillance scheme results including the treatment of uncertainties and consideration of any implications for the withdrawal scheme

- (GDA Issue GI-UKEPR-SI-02 “RPV Surveillance Scheme –Implications of Change in Neutron Energy Spectrum Caused by the Heavy Reflector ” http://www.onr.org.uk/new-reactors/reports/step-four/gda-issues/gda-issue-gi-ukepr-si-02.pdf)

New Build Plant – UK EPR™

- Subsequently, for GDA, EDF and AREVA proposed an approach which comprises:
  - Derivation of a tentative dose-damage correlation based on dpa as the dose parameter to take account of the differences in neutron spectrum between the EPR™ surveillance specimen location and the RPV wall.
  - Analyses of the advantages and disadvantages of fluence and dpa indexations.
  - An example of a flexible withdrawal scheme using dpa as the dose parameter

Conclusions

- UK Regulation is goal setting and not prescriptive.
- The ONR expectation for irradiation embrittlement of major vessels is that code compliance is not necessarily, in and of itself, sufficient.
- The UK has one Civil light-water reactor, Sizewell B, which has a surveillance programme containing Charpy and Fracture Toughness specimens.
- Operating Civil plant contains pre-strained samples; this was a direct result the UK Light-Water Reactor Study Group findings.

Conclusions

- New build reactors in the UK will demonstrate that they have suitable and sufficient surveillance programmes, going above code compliance and meeting UK ALARP expectations.
- The expectations vary between technologies and reactor designs, depending upon the nuclear safety significance, and the importance of the degradation mechanism to overall integrity.
- The requesting party must demonstrate an adequate understanding of the mechanism of irradiation embrittlement and how the specifics of the reactor design interact with this mechanism.

B.19 State of Knowledge and Research Activities on RPV Materials in UK (G. Burke)

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The University of Manchester
Dalton Nuclear Institute

State of Knowledge
and
Research Activities on RPV Materials in UK

Grace Burke

Director
Materials Performance Centre
University of Manchester

U.S. Nuclear Regulatory Commission International Workshop on
Age-Related Degradation of Reactor Vessels and Internals
23-24 May 2019

Materials Performance Centre

m.g.burke@manchester.ac.uk

MANCHESTER
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Dalton Nuclear Institute

Outline

- Background
- Evolution of current understanding of irradiation embrittlement of RPV materials
  - Properties and Physical Changes in Neutron-Irradiated Steels/Welds
  - Emphasis on model development (empirical and fundamental understanding) for predicting behavior / trends
  - Where are we now?
- Research activities in the UK (Irradiation-induced degradation of structural materials)
- Future

Materials Performance Centre

Irradiation Damage and Embrittlement of RPV Steels

- Mid- 1960's onwards: Unexpected shifts in ΔT_{41J} in weld surveillance specimens. Welds had been fabricated using Cu-coated weld wire. (SP Grant, 1968)
- PRE-DB development: CVN ΔT_{41J} data analysis leading to Reg. Guide development (and revisions)
- **UK (CEGB and UKAEA): Efforts on Magnox Steels and related welds (C-Mn steels not A533B/A302B or A508 Gr3)**
- **UK Dose-Damage relationships developed**

Irradiation Damage of Low Alloy Steels/Welds: Evaluation

- Late 1960's: Reports (S.P. Grant) of embrittlement in Cu-containing (~0.2-0.3 wt.% Cu) RPV Beltline weld specimens – concern for RPV structural integrity
- 1977: Development of US NRC Reg. Guide 1.99 (Cu and P)
  - Rev. 2 (Guthrie and Odette) for prediction of irradiation embrittlement (ΔT_{41J}) issued in 1988.

Irradiation Damage of Low Alloy Steels/Welds: Evaluation

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- 1977: Development of US NRC Reg. Guide 1.99 (Cu and P)
  - Rev. 2 (Guthrie and Odette) for prediction of irradiation embrittlement (ΔT_{41J}) issued in 1988.
- **1987 UK: Fisher and Buswell: Correlation of irradiation embrittlement data (ΔT_{41J}), steel composition and hardness**
  - **Assumed irradiation-induced Cu precipitation in ferrite**
  - **SANS data interpreted in terms of Cu precipitation + vacancies**
  - **Yield strength increase due to irradiation; linked to hardness increase.**

$$\Delta Hv_{irrad} = \Delta Hv_{matrix} + \Delta Hv_{Cu\ precipitation}$$

- **1986/88 UK: Williams *et al.*: Ni enhanced "Cu precipitation" in promoting embrittlement.**

Explanations call for justification based on microstructural characterisation

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Irradiation Damage/Embrittlement of RPV Steels (continued)

Research in 1970's+: Evaluating Cu effect on embrittlement (NRL and industry). Speculation of mechanisms. Conventional analytical techniques unable to identify the "changes" in material responsible for degraded properties. *(older methods lacked necessary resolution)*

- 1960's-1970's+: Analysis of materials/fundamental radiation damage studies/ "Russell-Brown model: Cu precipitation in Fe".
- 1983: Odette- Proposed formation of Cu-coated microvoids in welds (SANS);
- MTR irradiations (high Ni-Cu (Mn) welds); Odette: SMD, UMD, CRPs

Irradiation Damage/Embrittlement of RPV Steels (continued)

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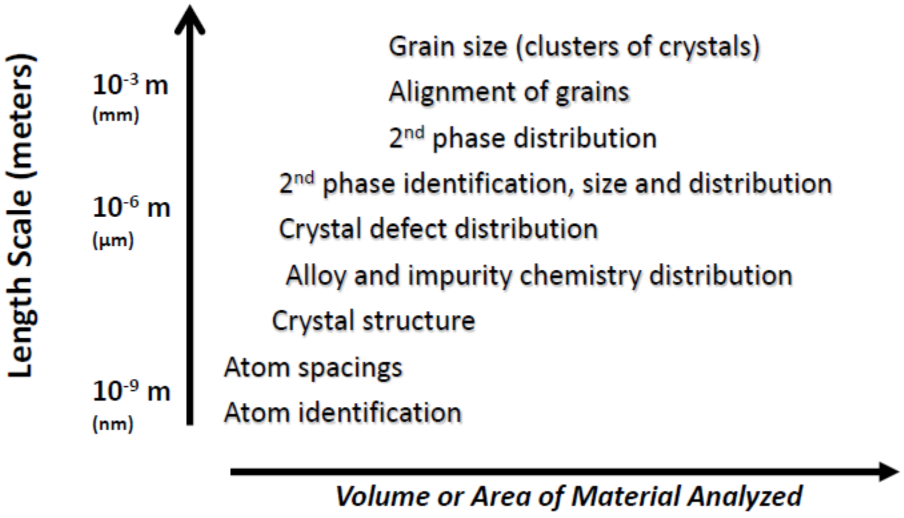
- 1983: Odette- Proposed formation of Cu-coated microvoids in welds (SANS);
- MTR irradiations (high Ni-Cu (Mn) welds); Odette: SMD, UMD, CRPs
- 1983-86: Burke, Brenner, Grant: first AP-FIM analysis of surveillance-irradiated high Cu (0.23 wt.%) and high Cu – high Ni welds; identified Cu-Ni-Mn-enriched solute clusters in welds (not visible by TEM or FIM).
- Subsequently numerous examples of irradiation-induced Cu-Ni-Mn(–Si)-enriched clusters from AP data from French (Rouen/EdF), US, UK, Japan (CRIEPI) during the 1990's-present.

Irradiation Damage of Low Alloy Steels/Welds: Microstructural Characterisation

Microstructural Analysis and empirical studies on irradiated commercial welds, steels and model steels needed complementary techniques (bulk and nanoscale analyses)

- TEM unable to detect features/defects in RPV surveillance steels/welds. *CIEMAT: demonstrated that defects could be imaged in model binary and ternary Fe-Mn-Ni, but not in commercial steels or welds.*
- **1980's-present:** AP-FIM and (1990's+) APT: continued research
- **Late 1990's – 2008:** Emphasis on combined techniques for characterization
  - APT/SANS/TEM/PIA-PALA-Hv of A508 Gr4N low Cu steels and welds (Burke *et al.)*
  - Ringhals weld studies (APT/SANS/HRTEM/PIA-Hv)
- **2013+:** Advanced ATEM providing new insights/assess variability/etc
  - Neutron-irradiated Welds/Steels
  - Ex-service components (pressurizer weld from Ringhals)

Different Instruments are Required to Resolve Fine and Coarse Microstructural Features



Grain size (clusters of crystals)

Alignment of grains

2nd phase distribution

2nd phase identification, size and distribution

Crystal defect distribution

Alloy and impurity chemistry distribution

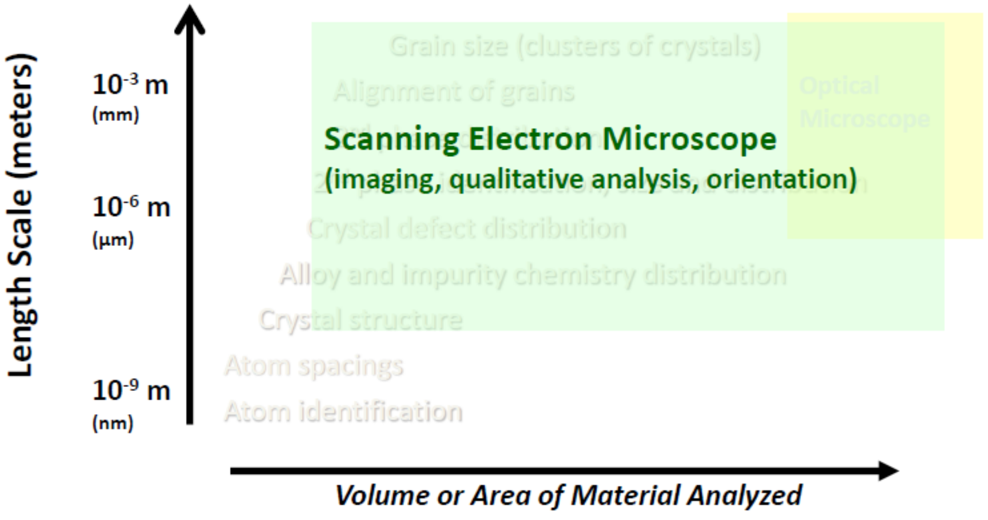
Crystal structure

Atom spacings

Atom identification

Volume or Area of Material Analyzed

Different Instruments are Required to Resolve Fine and Coarse Microstructural Features



Grain size (clusters of crystals)

Alignment of grains

Scanning Electron Microscope

(imaging, qualitative analysis, orientation)

Crystal defect distribution

Alloy and impurity chemistry distribution

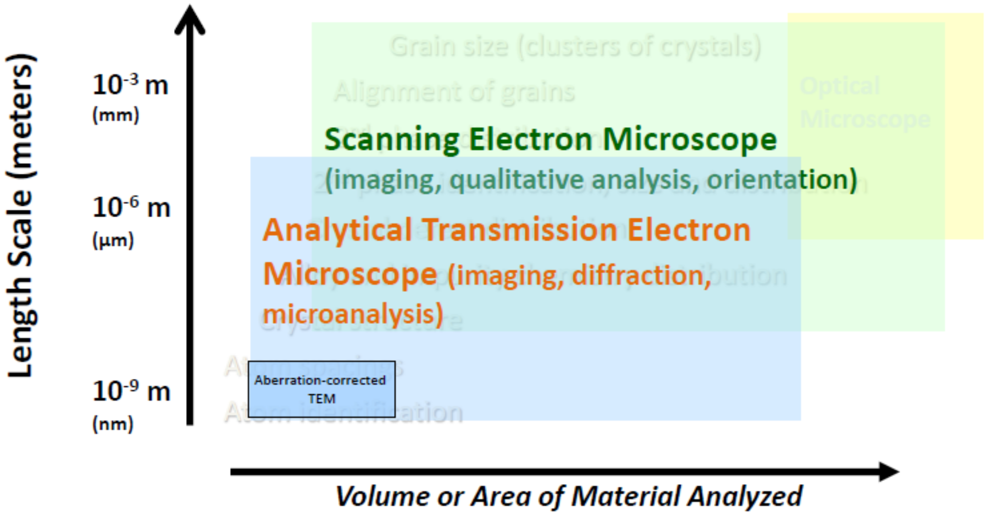
Crystal structure

Atom spacings

Atom identification

Volume or Area of Material Analyzed

Different Instruments are Required to Resolve Fine and Coarse Microstructural Features

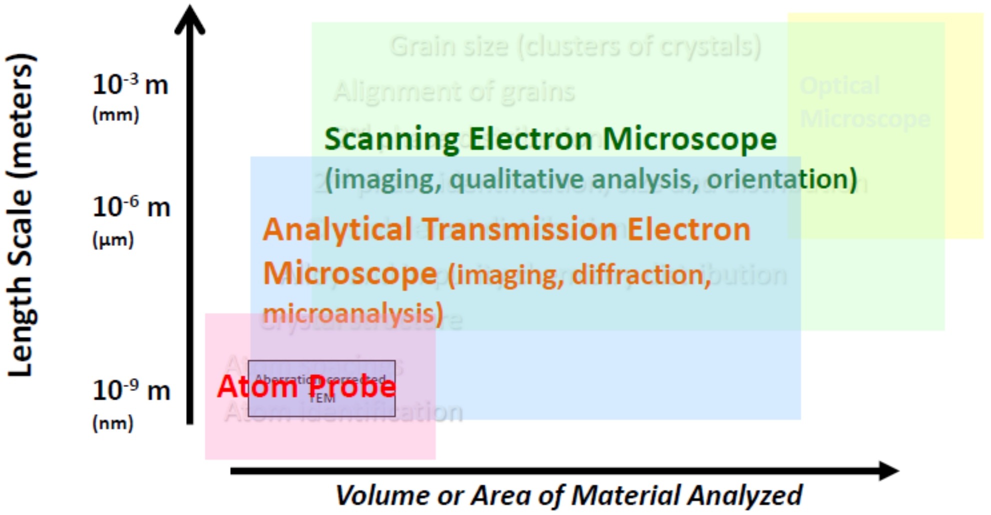


Scanning Electron Microscope (imaging, qualitative analysis, orientation)

Analytical Transmission Electron Microscope (imaging, diffraction, microanalysis)

Materials Performance Centre

Different Instruments are Required to Resolve Fine and Coarse Microstructural Features



Scanning Electron Microscope (imaging, qualitative analysis, orientation)

Analytical Transmission Electron Microscope (imaging, diffraction, microanalysis)

Atom Probe

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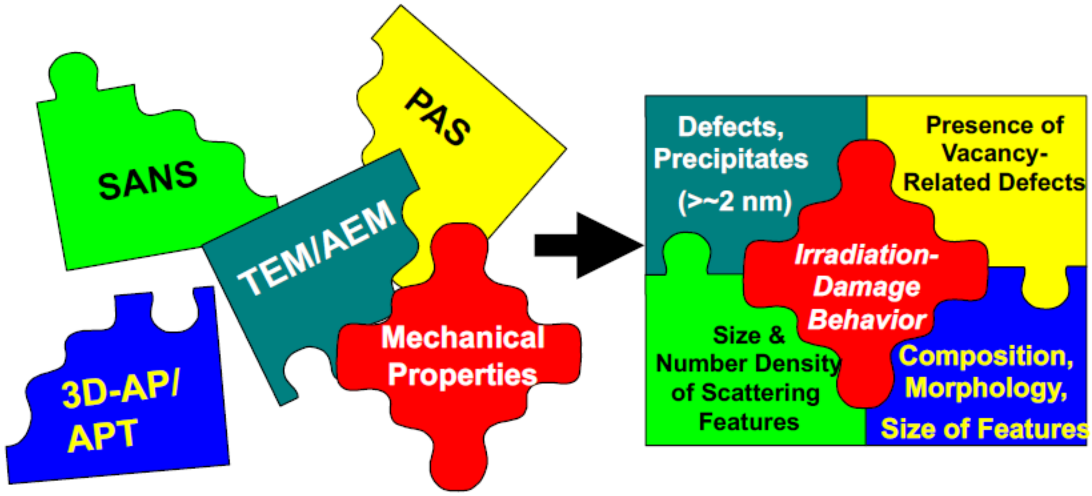
Dalton Nuclear Institute

Irradiation Embrittlement of RPV Steels:
→ Identifying the Physical Changes



SANS

PAS

TEM/AEM

3D-AP/ APT

Mechanical Properties

Defects, Precipitates (>~2 nm)

Presence of Vacancy-Related Defects

Irradiation-Damage Behavior

Size & Number Density of Scattering Features

Composition, Morphology, Size of Features

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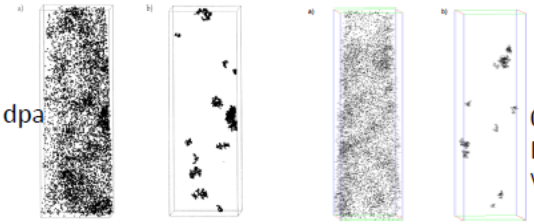
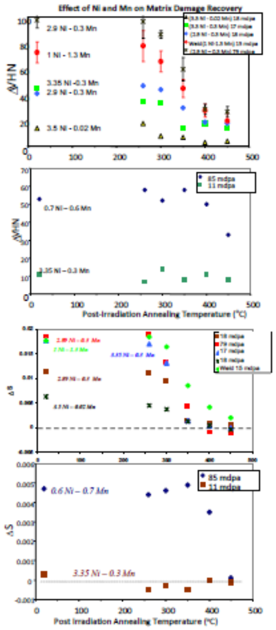
A508 Gr4N Research into Irradiation Damage Mechanisms (1998-2003)

Atom probe data show solute clustering following hardening trend



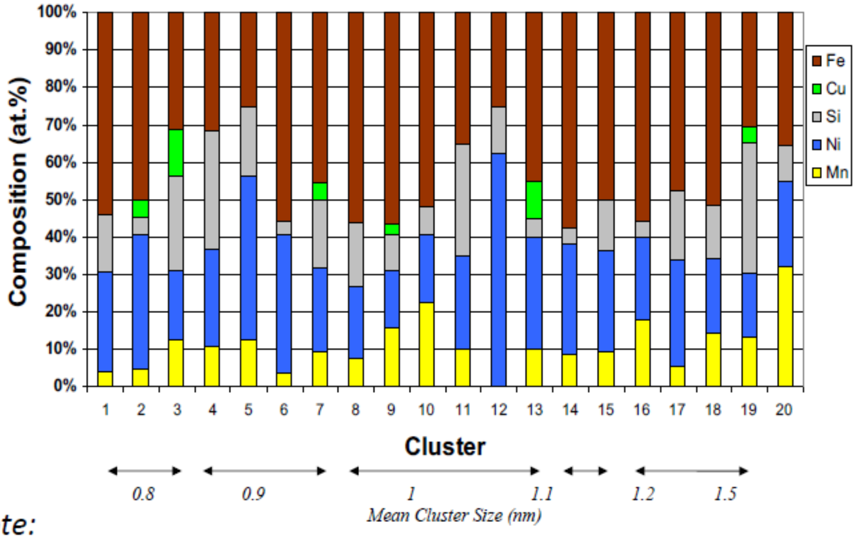
0.79 dpa

0.79 dpa + PIA (remove Vacancy damage)



Mn controls the "Ni Effect" in enhancing embrittlement in high Ni steels

1 Ni – 1.3 Mn - 0.03 Cu Weld: 95 mpda
APT Solute Cluster Composition (at.%)



Note:

- All solute-enriched clusters contain Ni and Si, all but 1 contain Mn.
- Only ~30% of the clusters contain Cu
- No effect of size on composition

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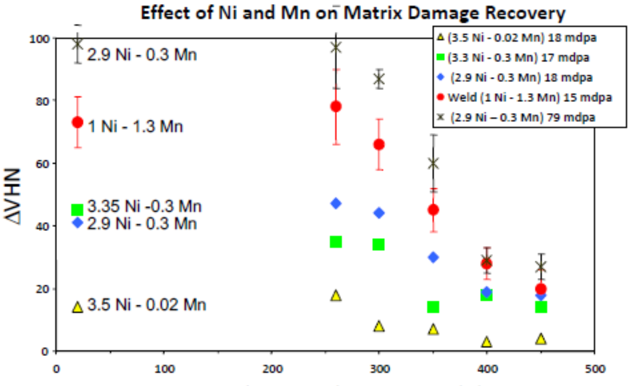
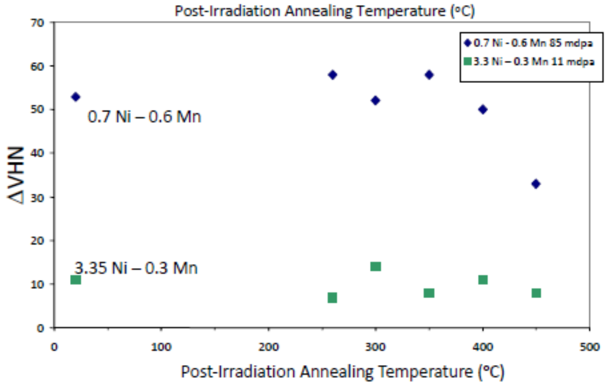
Comparison of High and Low Flux Irradiation Damage on ΔVHN

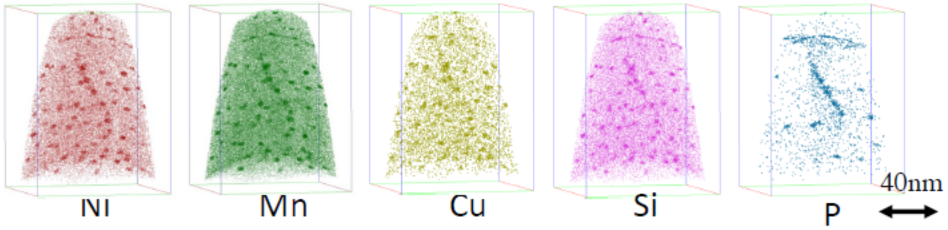
Note “less” irradiation-induced hardening damage in low flux 3.35 Ni **– 0.3 Mn** sample and “less” hardening in 3.5 Ni - **0.02Mn** high flux sample

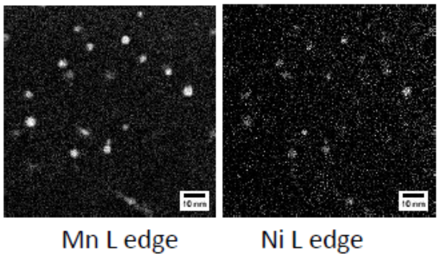


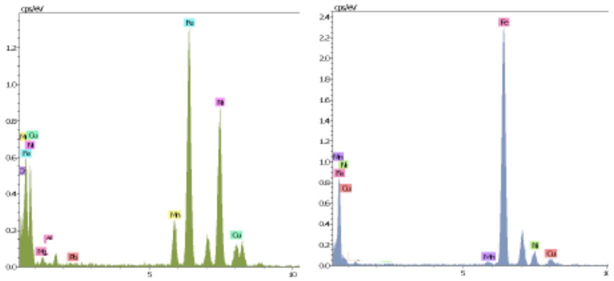


Ringhals 1.6 Ni – 1.4 Mn – 0.05 (and 0.08) Cu Welds Surveillance-Irradiated (0.073 dpa)

- 3-4 nm irradiation-induced Ni-Mn-Si-(Cu)-enriched clusters
- Solute-decorated dislocations



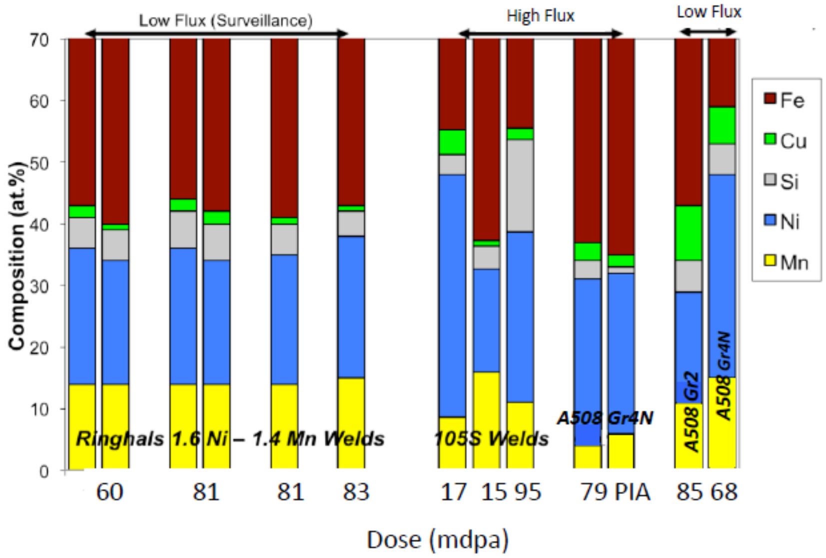




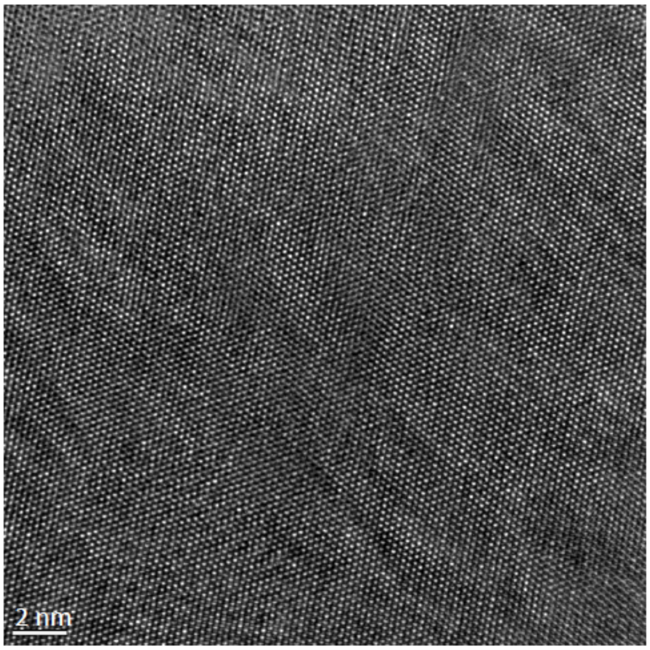
Summary of 3D-AP Data

High Ni Welds and Ni Steels (Low and High ϕ) show consistent solute clusters

Average Cluster composition (at.%)



Aberration-Corrected TEM
Irradiated Weld: 6.7 X10^{19} n/cm^2



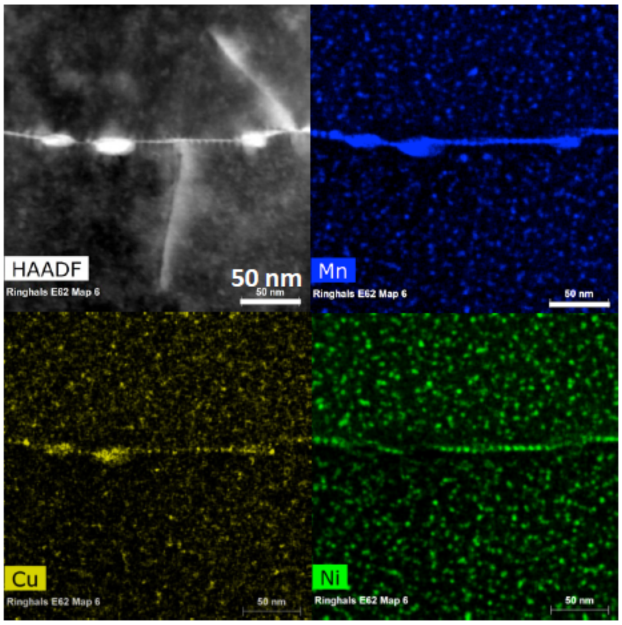
[111] high resolution TEM image

No clearly visible precipitates

"New" Technique: Advanced Analytical S/TEM

Ringhals 1.6 Ni – 1.4 Mn – 0.08 Cu Weld (0.073 dpa)

- 2-5 nm irradiation-induced Ni-Mn-Si-(Cu)-enriched clusters
- Cu-Mn ε precipitates
- Large areas of analysis
- STEM-EDX SI – all elements collected permitting subsequent reanalysis
- Excellent agreement with APT data

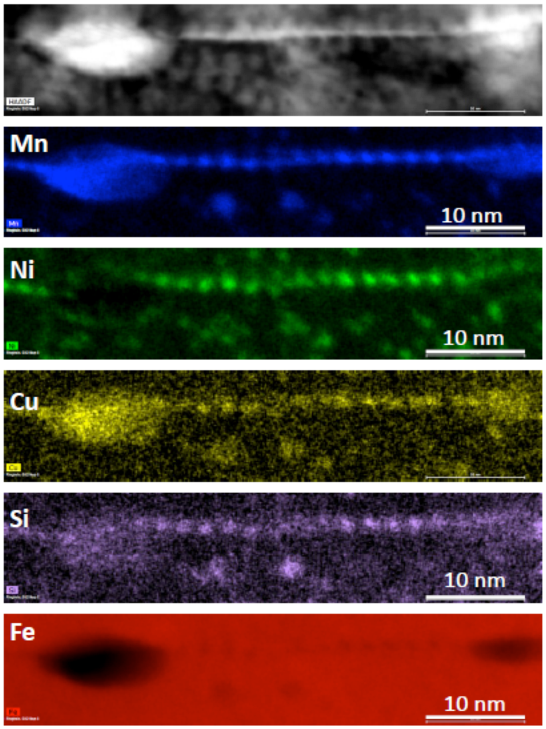


”New” Technique: Advanced Analytical S/TEM

Talos STEM-EDX Spectrum Image Data

Cu-Mn (epsilon Cu) – like precipitates consistent with previous studies on Cu-containing RPV welds

Note discrete solute-enriched clusters along boundary and within matrix



EDX elemental maps

RPV Steels and Welds

→ *Continuum of solute-related hardening features in contrast to single Cu precipitation term.*

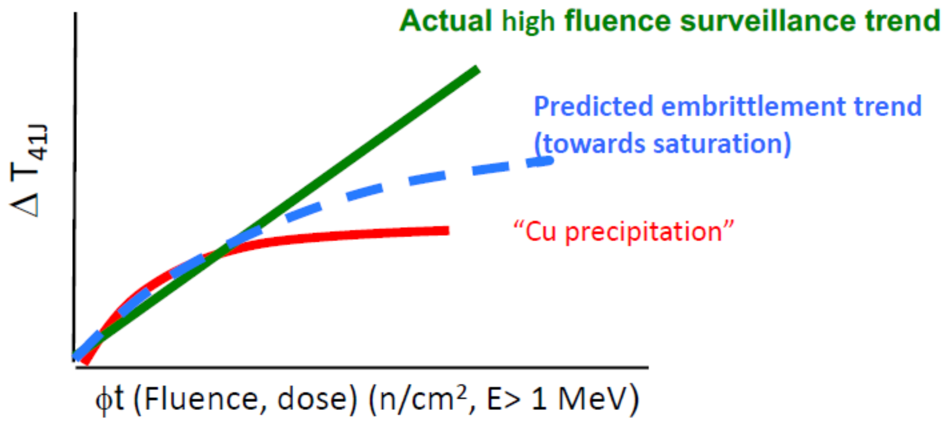
From Irradiation Damage Mechanism studies of high Ni steels and welds

$$\Delta \mathbf{Hv}_{\text{irrad}} = \Delta \mathbf{Hv}_{\text{vac(PALA)}} + \Delta \mathbf{Hv}_{\text{sol(AP)}}$$

Data consistent with Russell-Brown hardening (for measured solute-related hardening)

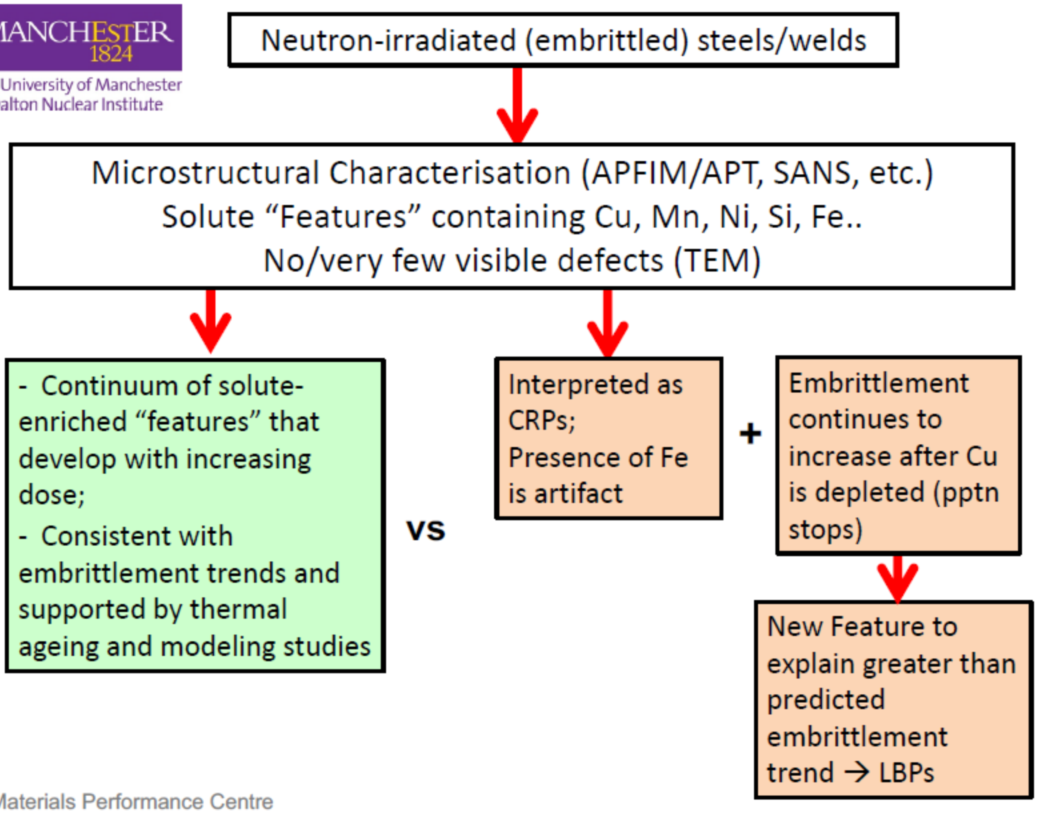
No need to invoke so-called “Late Blooming Phases”/ Ni-Mn precipitates/ phases/etc.

Irradiation embrittlement can be explained by the evolution of solute-enriched clusters…



- Presence of Mn and Ni (- Si) in clusters from inception – presence of Fe is not an artefact.
- No need to invoke precipitation and Cu exhaustion, i.e. LBPs, to explain high fluence trends

Materials Performance Centre



Neutron-irradiated (embrittled) steels/welds

Microstructural Characterisation (APFIM/APT, SANS, etc.)
Solute "Features" containing Cu, Mn, Ni, Si, Fe..
No/very few visible defects (TEM)

- Continuum of solute-enriched "features" that develop with increasing dose;
- Consistent with embrittlement trends and supported by thermal ageing and modeling studies

vs

Interpreted as CRPs; Presence of Fe is artifact

+

Embrittlement continues to increase after Cu is depleted (pptn stops)

New Feature to explain greater than predicted embrittlement trend → LBPs

Materials Performance Centre

So….. The appears to be a consistent picture concerning irradiation-induced “solute-related” hardening “features” based on numerous RPV and LA Steels/Welds …

- Can these observations be predicted from atomistic modelling?
- Can atomistic modelling help to interpret our microstructural / microchemical observations – particularly with respect to solute-enriched hardening features and vacancy-related hardening?
- EU coordinated programme activities in PERFORM 60, SOTERIA and national programmes

EU PERFORM60 & LONGLIFE Activities

- Successfully developed interatomic potentials necessary to address the complex Cu-Mn-Ni-Si-Fe interactions (*Becquart, Malerba and colleagues*)
- Demonstrated vacancy-Mn (and other significant solute interactions) binding; independent confirmation of experimentally-measured irradiation damage behaviour of production high Ni steels with/without Mn (A508 Gr4N)
- Demonstrated experimentally and via modeling the formation of irradiation-induced defects (effect of alloying on TEM imaging/ visibility) from binary/ternary model alloys to RPV steel. (*Mayoral, Malerba*)
- Demonstrated that very long-term ageing at 365°C can produce CuMnNiSiFe “precipitates” (thus lending support for the irradiation-induced solute-enriched clusters formed due to neutron irradiation *(Styman et al.)*

On-going Research

- Issue: Degradation of materials in reactor environments….
  - Changes in material properties as a function of:
    - * neutron irradiation
    - * environment (non-neutron contribution)
- Identification of physical changes in material associated with degradation in materials performance…
  - Detailed characterisation of the microstructure to identify those changes that are associated with changes in material properties/performance
- Development of models to predict material performance
- Development of improved materials/processing/heat-treatments and component design for existing and new reactors

UK National Nuclear Users Facilities (NNUF)

UK Government investment in facilities for nuclear R&D

NNL Central Laboratory (Sellafield) *(high activity work)*
UKAEA Materials Research Facility (Culham)* *(lower activity)*
Culham Centre for Fusion Energy (Culham).
Dalton Cumbrian Facility for Proton and Ion Irradiation
University of Huddersfield MIAMI-1 and MIAMI-2 TEMs for *in situ* ion irradiations (Microscope and Ion Accelerator for Materials Invesitgations)

Linked with Nuclear Advanced Manufacturing Research Centre (U Sheffield) and the new Henry Royce Institute for Advanced Materials (active laboratories for Graphite and Fuels at the University of Manchester)

Lack of MTR facilities….

Can proton irradiation be used to produce similar damage forms in low alloy steels

Potential for Research Applications?

Inexpensive
Fast
Materials can be readily handled (no hot cell required)
X Protons are not neutrons!
X Limited penetration depth (~15-~40 microns)

UK Dalton Cumbrian Facility for Ion/proton irradiation
Variables: dose and T_{irrad}

Ongoing Research into High Toughness A508 Gr4N Forging Steels

- High Ni (~3.3-3.7 wt%) and low Mn (**0.02** to 0.3 wt%)
- Martensitic-Lower Bainitic Steels – As-cooled + Tempered
- Outstanding toughness
- Irradiation Damage Behavior?
  - Comparable to lower Ni steels – No “enhanced embrittlement”
  - Extensive multi-technique research program (1998-2003) (APT, TEM, HRTEM, SANS, PALA, PIA) demonstrated the **critical role of Mn** in the development of stable, neutron-irradiation-induced hardening
- Can we generate similar hardening and structures using proton irradiation?

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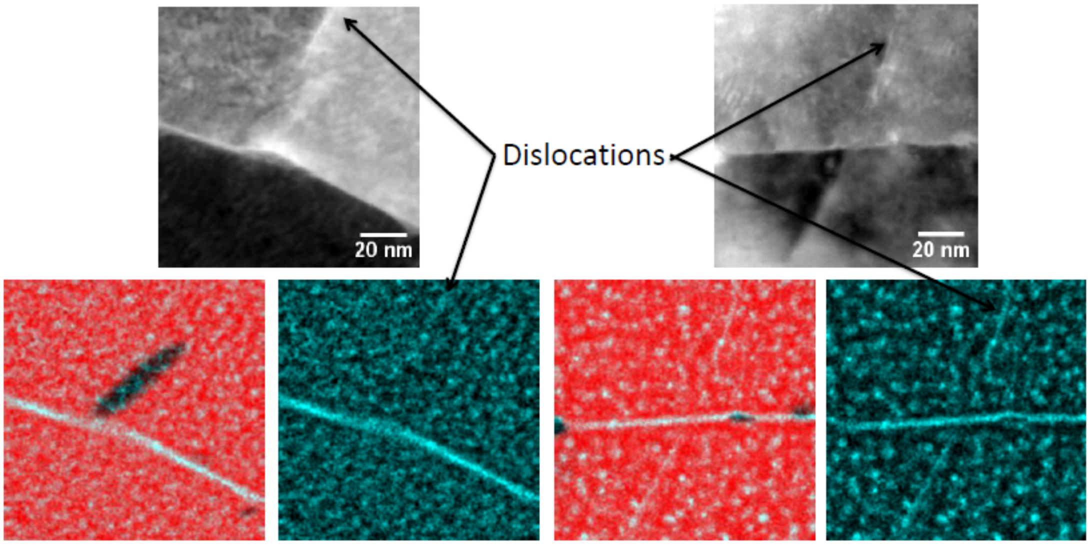
A508 Gr4N Steel
Neutrons vs. Protons (~225°C)

Neutron Irradiation 0.092 dpa **Proton Irradiation 0.4 dpa**



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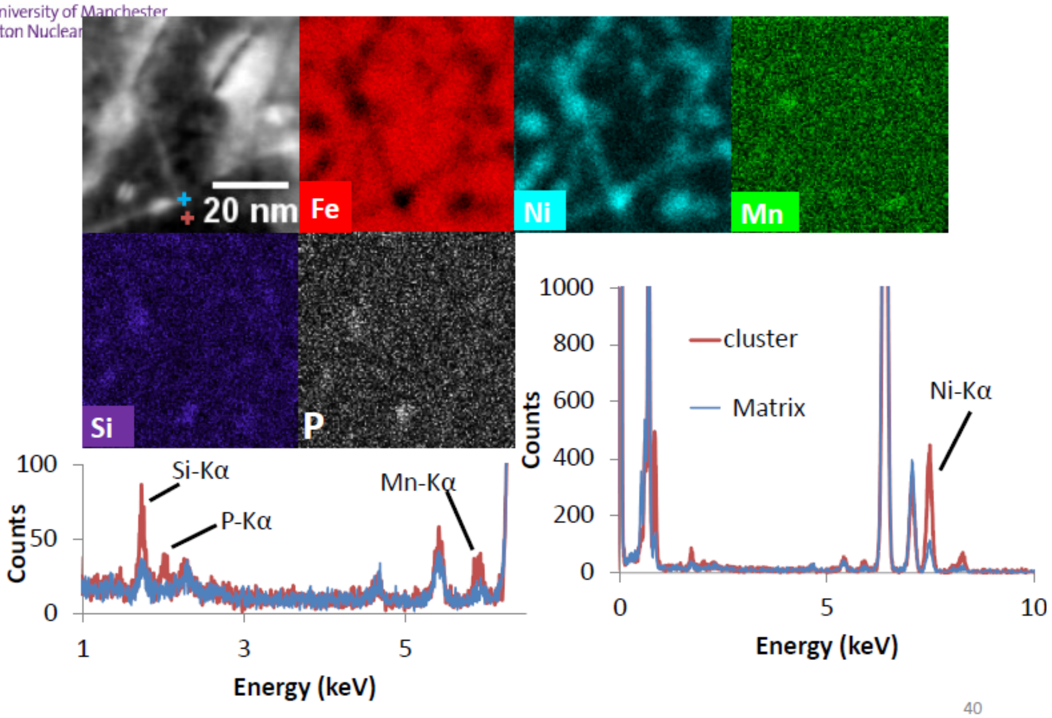
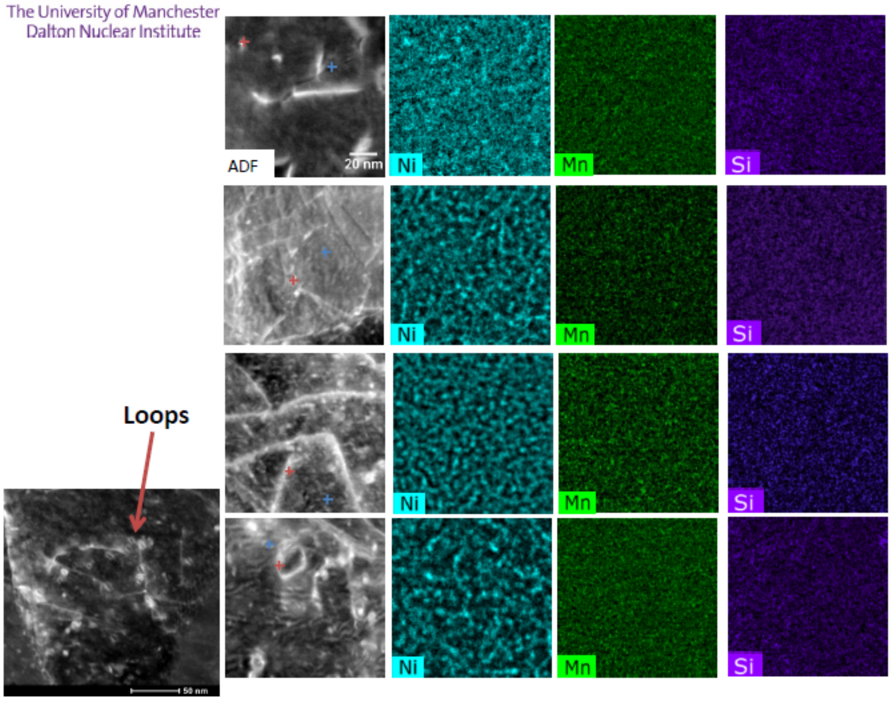
Proton Irradiation at 320 °C: 1.1 dpa



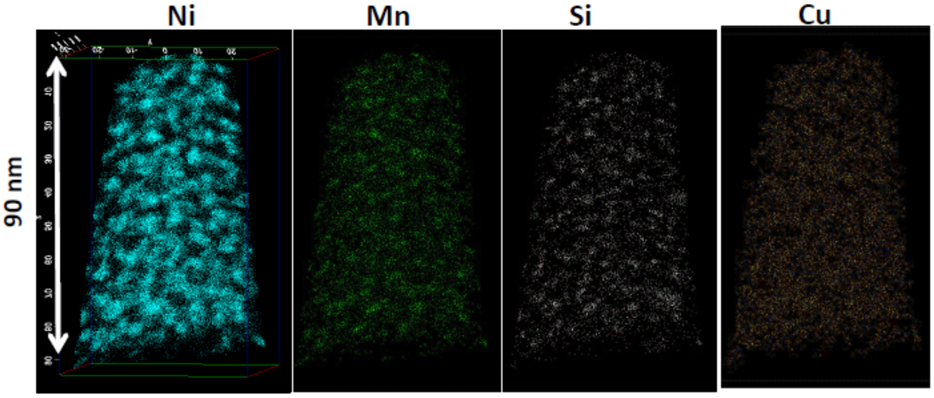
Proton Irradiation: Effect of Dose and T_{irrad}



T (°C)	dpa
225	0.09
225	0.4
225	1.1
320	1.1

But – how do Advanced ATEM STEM-EDX SI data compare with APT data?

0.4 dpa – 225°C Proton-Irradiated A508 Gr4N APT Results

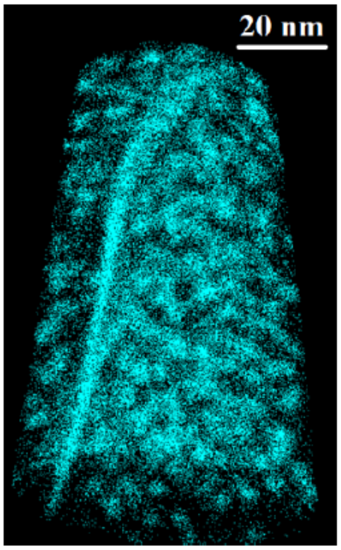


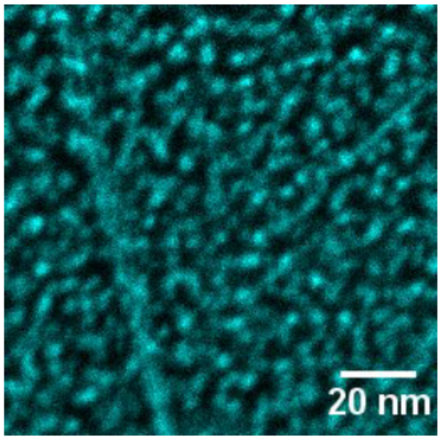
Collaboration with James Douglas/Michael Moody (Oxford)

Solute Clustering (Ni) in A508 Gr4N Steel Induced by Proton Irradiation

Comparison of APT and Talos STEM-EDX SI Data

0.4 dpa at 225° C





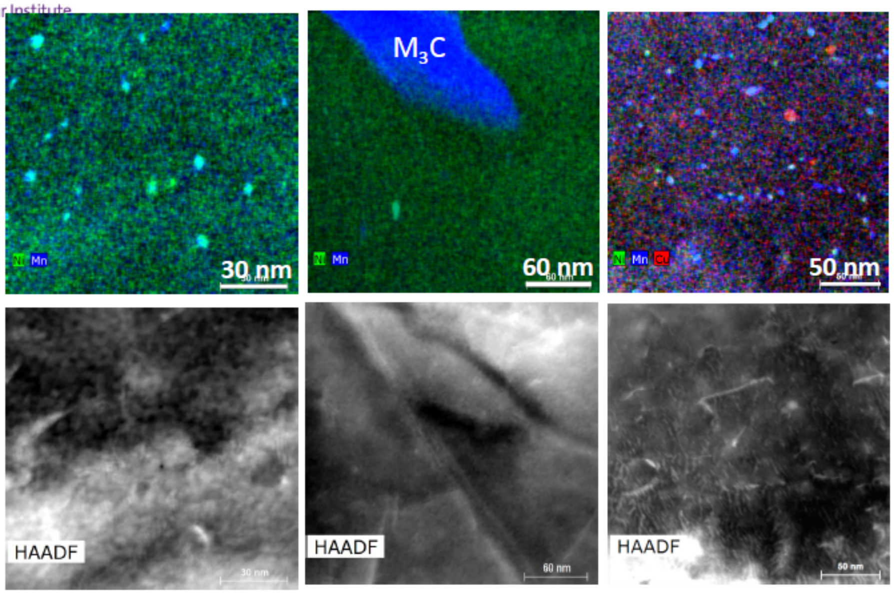
Collaboration with James Douglas/Michael Moody (Oxford)

Thermal Contribution to Neutron-Irradiated RPV Steels/Welds?

(Current Research - Vattenfall)

- Understanding role of fluence (dose) and flux (dose rate) on microstructural development/evolution of solute-enriched hardening “features” (clusters)
- Thermal contribution?
- Pressurizer: Higher temperature (~345ºC) than RPV (~300-310ºC) for 28 years
- **No increased hardening in base steel**….increase in hardness for weld and increase in ΔT_{41J} ***(less than neutron-irradiated)***

Pressurizer Weld – ~28 years at 345° C
FEG-STEM EDX SI Datasets



- **Non-uniform** nanoscale precipitation enriched in (Ni-Mn-(Cu))
- Preferential sites: GBs, dislocations

Lim, Burke, Efsing, To be published

Status

Multi-Scale Data Enables Fundamental Understanding and Mechanistically-based Predictive Models

- Explains Hardening/Mech Properties
- Key "Features" Identified: Size/Composition – Vacancy-related Defects ("Open Volume Defects")
- ***Assessing Variability – Importance of "Baseline" Properties*** (Microstructural Variability Documented)
- **Data Analyses must be SELF-CONSISTENT and CONSISTENT** with independent Complementary Analyses (Multi-Technique approach)
- **More Microstructure data needed for model development**

Route Forward? Coordinated/Collaborative Programs

Future Needs

- **Extracting data from real components to assess performance:**

Contributions of thermal ageing (pressurizer weld/pressurizer – note that T is ~40 C-deg higher than operation, but same/similar materials. Note : No shift for ex-service plate but weld showed hardening and a measurable shift. (As T decreases, thermal contribution is decreased).

- **SMR Development/Materials Validation:** Need for lower T data (K and ΔT_{41J})

How to generate?

Existing support structures?

Low flux and low T_{irrad}. Need both mech property data and analysis of material (determine physical changes in microstructure) for input to predictive model(s).

- Benefit of IGRDM (originally started with US NRC guidance)

Need for collaborative, jointly-sponsored international research programs.

Acknowledgements

EPSRC - EP/N017854/1

UKAEA MRF

UK National Nuclear Laboratory

ROLL-ROYCE

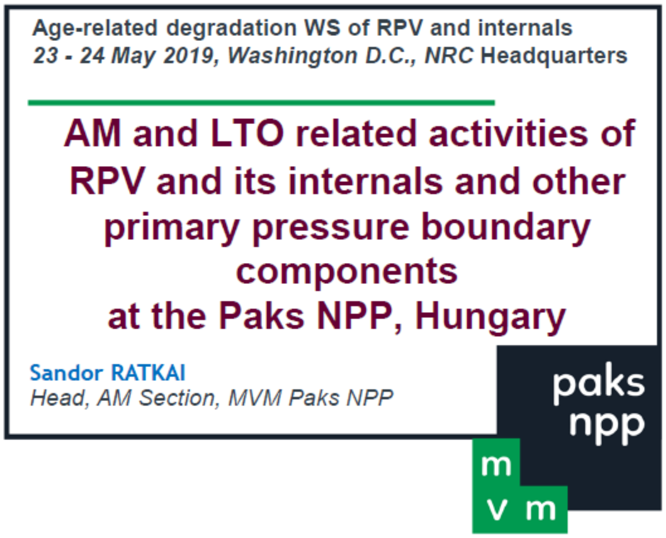
VATTENFALL

Horizon 2020

Many colleagues in the US and UK (MRF, MPC, Oxford University)

Special acknowledgement to the late S.P. Grant

B.20 AM and LTO-Related Activities of RPV and Its Internals and Other Primary Pressure Boundary Components at the Paks NPP (S. Ratkai)



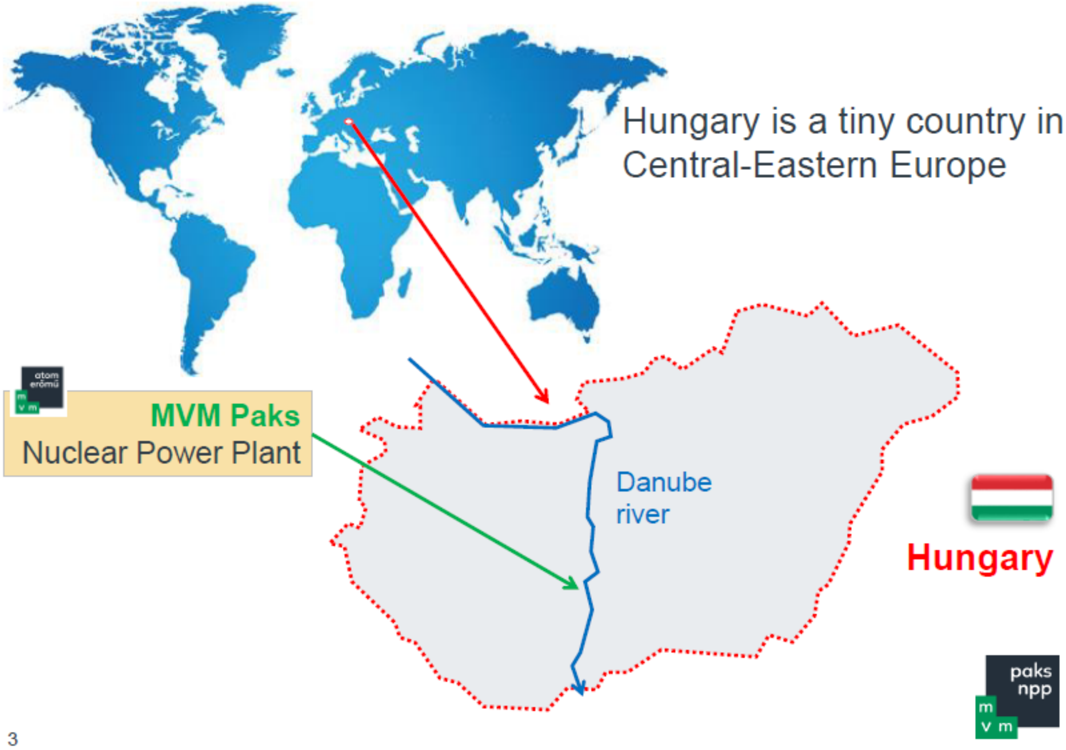
Age-related degradation WS of RPV and internals
***23 - 24 May 2019, Washington D.C., NRC* Headquarters**

AM and LTO related activities of RPV and its internals and other primary pressure boundary components at the Paks NPP, Hungary

Sandor RATKAI
Head, AM Section, MVM Paks NPP

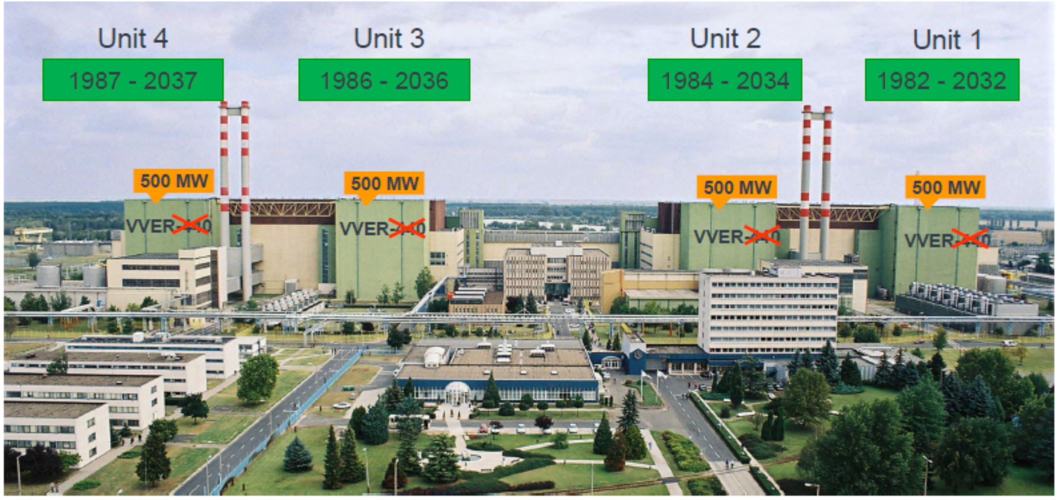
Content

Hungary and MVM Paks NPP



Hungary is a tiny country in Central-Eastern Europe

LTO at MVM Paks Nuclear Power Plant (50 years)



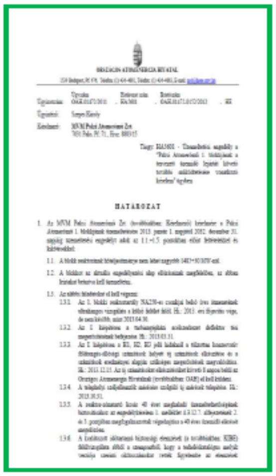
VVER440/V213 units *(second generation)*

- ***Russian design of PWR (VVER) (twin Units)***
- ***Power uprates conducted for 500 MW electricity output***
- ***Original design life is 30 years***
- ***Target operating life is 50 years (LTO: 30+20 years)***

Long Term Operation of Paks Units

Units	Start-up	Design operating life (30 years)	Extended operating licence (30 + 20 years)
Unit 1	1982	2012	**2032 ✓**
Unit 2	1984	2014	**2034 ✓**
Unit 3	1986	2016	**2034 ✓**
Unit 4	1987	2017	**2037 ✓**

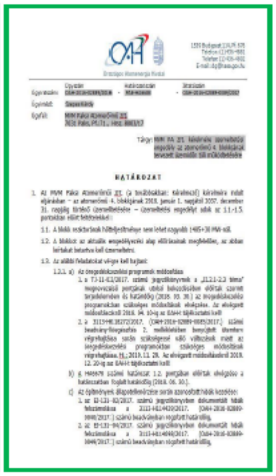
Extended Operation Licenses (HAEA resolutions)



HA 5601 – Unit 1

HA 5923 – Unit 2

HA 6485 – Unit 3



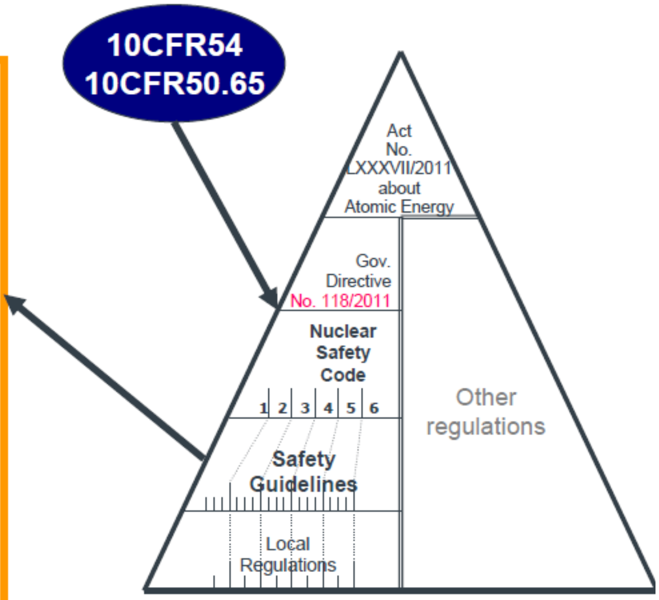
HA 6688 – Unit 4

Regulatory and technical approaches for LR and Ageing Management

Nuclear regulation relative to LR

Guideline 1.28
Regulatory procedures on the operation beyond the design lifetime

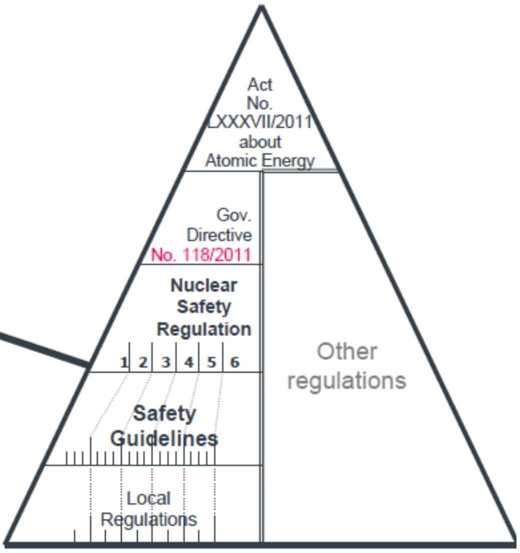
Guideline 4.14
Activities to be implemented by the operator to support the License Renewal application for operation beyond design lifetime



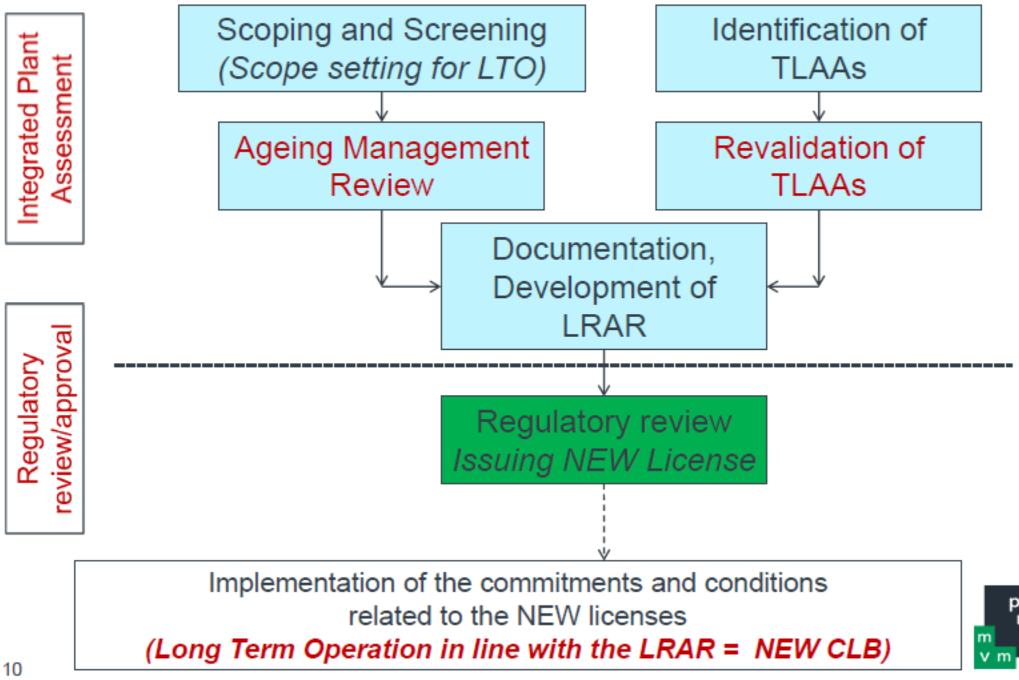
Nuclear regulation relative to AM

Guideline 3.13
Consideration of ageing mechanism during the design

Guideline 4.12
Ageing management during the operation of NPPs



Technical basis of LR



Scoping and Screening
(Scope setting for LTO)

Identification of TLAAs

Ageing Management Review

Revalidation of TLAAs

Documentation, Development of LRAR

Regulatory review
Issuing NEW License

Implementation of the commitments and conditions related to the NEW licenses
(Long Term Operation in line with the LRAR = NEW CLB)

Scope setting and screening for AMR

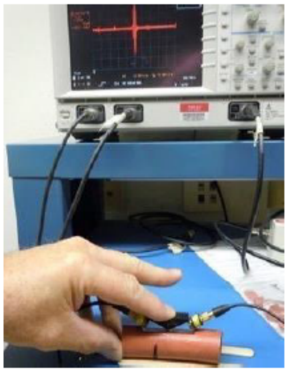
Scope setting:

- Safety Class 1-3+ (~ 25000 SCs/ Unit);
- Justification of the completeness;
  (diagrams + data base);
- Reproducibility;
- Listed by:
  - ➢ diagram based – mechanical components;
  - ➢ building based – civil structures;
  - ➢ diagram based – I&C components;

Screening for AMR:

- Long lived Passive SCs;

Ageing Management Programmes

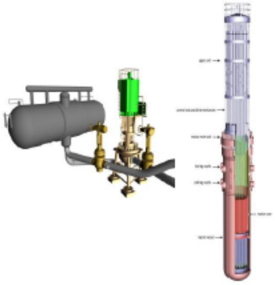


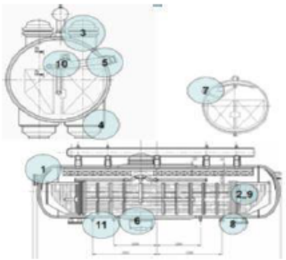
Main features:

- Based on the US NRC GALL, and updated by the IAEA IGALL
- ~150 AMPs were developed for passive safety related components (component or commodity groups oriented)
- Active components are managed by the Maintenance Effectiveness Monitoring (*MEM)*
- Very detailed programmes, 50 to 150 pages each
- Includes references to other operation AM programmes (e.g. ISI, water chemistry, condition oriented programmes, etc..)
- **Living programmes (regularly updated according to the new R&D results and/or OPEX e.g. US (EPRI), EU (NUGENIA)**
- **AM related research needs were identified and those are consequently performed**

Example for the content of AMP: Steam Generator

(Attribute 1 - ageing effects/degradation mechanism and critical locations)





Degradation mechanism / Location	Fatigue	General corrosion	Boric acid induced corrosion	Local corrosion (stress corrosion)	Wear	Loss of preload	Deposit
Casing/welds/nozzle areas		+	+	+			+
Heat-exchanging tubes				+			+
Connections of the collector covers/other flanged joints /bolted connections	+	+		+	+	+	
Primary circuit collectors	+			+			
Feedwater inlet nozzle		+		+			
Connection areas of the NA 500 main circulation pipeline nozzles with dissimilar welds				+			+
Inlet nozzle of the emergency feed-water		+		+			
Directly connecting supporting structures, the earthquake protection reinforcements		+	+	+	+	+	
Heat exchanger pipe plugs and welds				+			
Feed-water collector				+		+	
Blow-down nozzles, cleaning nozzles		+		+			+

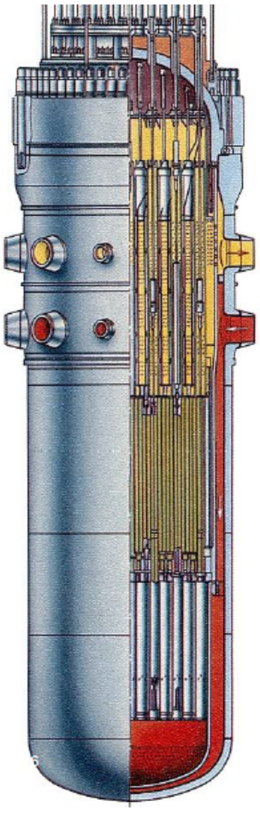
Time Limited Ageing Analyses

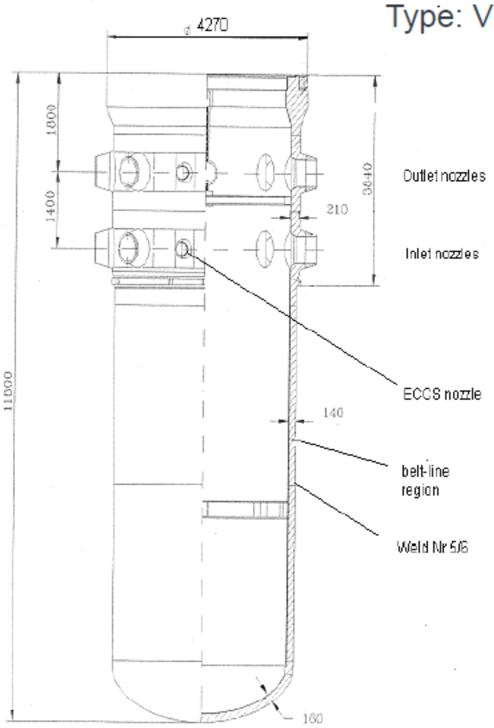
- 30 calculations were identified;
- New calculations were performed;
- State-of-the-art methods were applied:
  - most important (Class 1-3) **mechanical components:** design review has been performed taking the ASME BPVC III requirements;
  - all **building structures** were checked against the EUROCODE;
  - all safety related **I&C components** were subject to EQ (IEEE 323 or IEC 60780 & 60502);

Main TLAA of RPV
Pressurized Thermal Shock (PTS)

Reactor Pressure Vessel

Type: VVER440/213



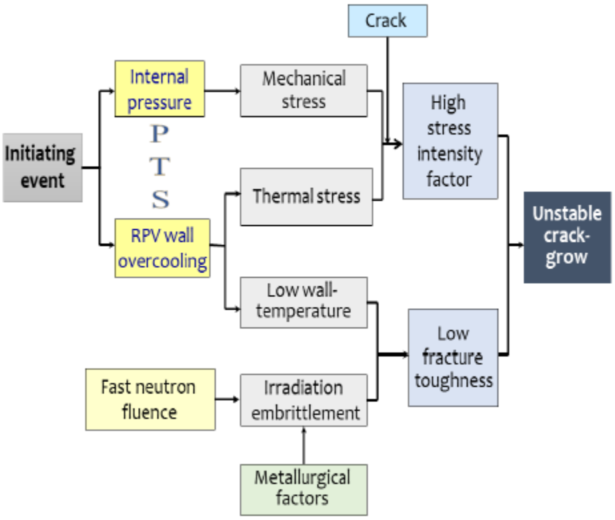


Original Design Parameters

Mass	215 t
Length	11,800 m
Diameter (cylindrical region)	3,840 m
Diameter (nozzle region)	3,980 m
Wall thickness (cylindrical region)	0,140 m
Wall thickness (nozzle region)	0,210 m
Number of nozzles	2 x 6 (primary) + 2 x 2 (ECCS)
Operational pressure	12,26 MPa
Design pressure	13,7 MPa
Hydrotest pressure	19,12 MPa (original) 16,64 MPa (since 1992)
Operational temperature	265 °C
Design temperature	325 °C
Design life	40 year
End-of-life fluence (base metal)	2,6 x 10^{24} $(E > 0{,}5 MeV)$
End-of-life fluence (Weld No 5/6)	1,8 x 10^{24} $(E > 0{,}5 MeV))$

Pressurized Thermal Shock – process and impact

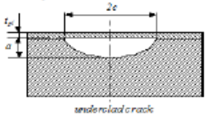
- **PTS: overcooling transient** causing thermal shock to RPV wall
- Pressure is either maintained or the system is re-pressurized
- Thermal stress in combination with pressure stress results in large tensile stresses (maximum: in inside surface)
- **Irradiation embrittlement (environmental effect)**, occurs in the RPV wall reducing fracture toughness and shifting transition temperature to higher temperature
- If a crack exists near to inside surface, where the material degraded due to irradiation, and a PTS transient happens, the RPV integrity is jeopardized



Elements of PTS analysis

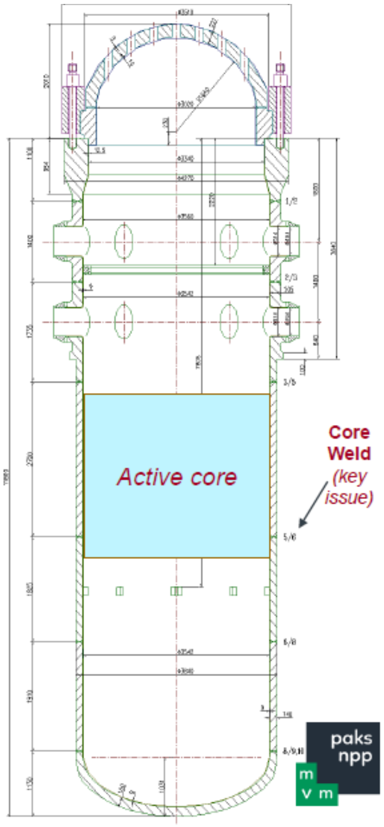
- 50+10 operation years
- PTS transient identification
  - deterministic (based on engineering judgement)
  - probabilistic (PSA based) – f>10^{-5}/year
- Thermal-hydraulic calculations
  - RELAP5/mod3.2, ATHLET
  - flow stagnation: REMIX
- Fast neutron fluence calculations
  - KARATE and MCNP for RPV wall and surveillance position
- Evaluation of irradiation effects on RPV materials
  - surveillance programmes (see those later)
  - trend curve
- Evaluation of in-service inspection results
- Postulated cracks: semi-elliptic underclad crack set

a = 0.1 t, a/c = 1/3



- Structural integrity analyses (LE FM, EP FM) ➡ Allowable transition temperature: T_k^{allow}

T_k^{allow}: 5/6 welding=166,7 [°C]; forging (at core)=200,4 [°C]



Core Weld *(key issue)*

Surveillance programmes of RPV

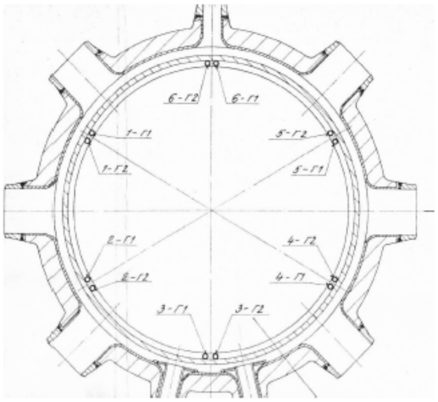
Features of the surveillance programmes

- Original programme: „Loviisa” type
- New programmes: Hungarian designed (by Hungarian research institutes)
- Inspection in hot chambers at the Paks NPP
- Evaluation by Hungarian research institutes and universities
- Independent controls:
  - Expert panel of RPV (1984 – 1995, 2003 - 2008)
  - Expert panel of Structural Integrity (2008 -)

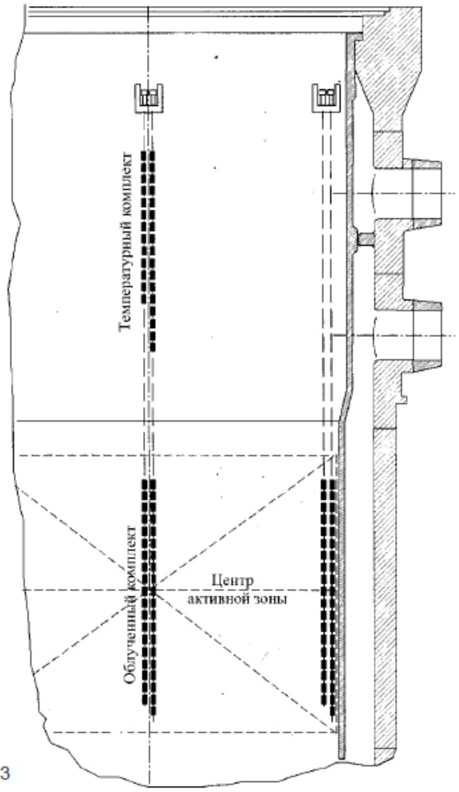
Original surveillance programme

VVER 440/213 surveillance system:

- 6 complete specimen sets were prepared at the factory (base metal, weld and HAZ material)
- 5 specimen sets were tested in the frame of the surveillance programme during the first 4 campaigns (lead factors: 12 (base metal), 16 (5/6 welding)
- serves until the original design life (30 years)



Specimens

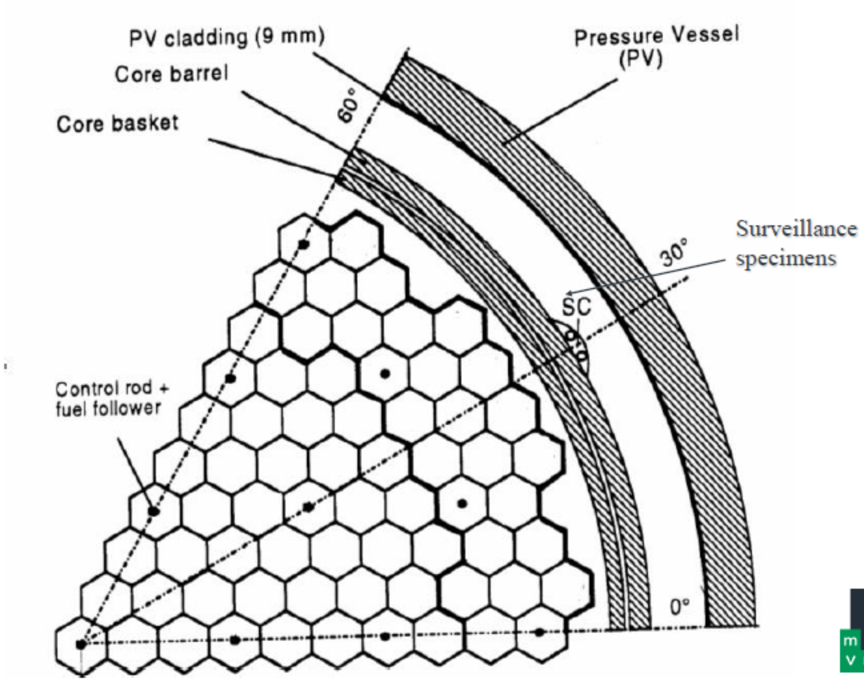


Type of the specimens:
- tensile test specimens
- *Charpy* impact specimens
- fracture mechanics specimens
  [Three Point Bending (TPB/COD)]
- neutron monitors
- temperature check monitors
  (diamond powder)

Material of the specimens:
- base material
- welding material
- heat affected zone material

Condition (heat treatment):
- same as the RPV

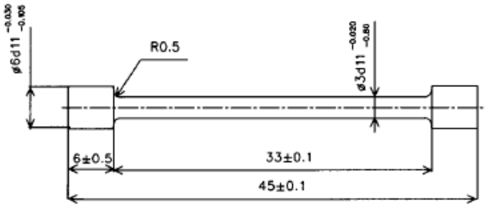
Surveillance test kits (VVER-440/213)



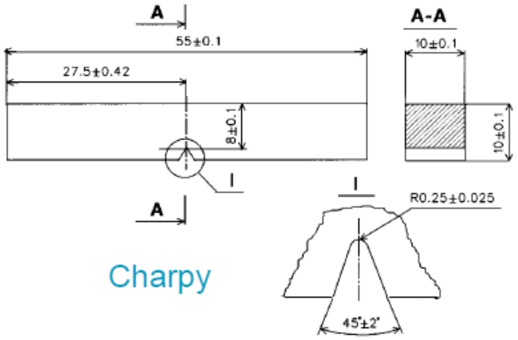
Summary of the specimens

Set number		Chain number	Base metal			Weld metal			HAZ		
			Specimens			Specimens			Specimens		
			Charpy	TPB	Tensile	Charpy	TPB	Tensile	Charpy	TPB	Tensile
Opposite the core	1	1-1	12	12	6	-	6	-	-	-	-
		1-2	-	-	-	12	6	6	12	12	6
	2	2-1	12	12	6	-	6	-	-	-	-
		2-2	-	-	-	12	6	6	12	12	6
	3	3-1	12	12	6	-	6	-	-	-	-
		3-2	-	-	-	12	6	6	12	12	6
	4	4-1	12	12	6	-	6	-	-	-	-
		4-2	-	-	-	12	6	6	12	12	6
	5	5-1	12	12	6	-	6	-	-	-	-
		5-2	-	-	-	12	6	6	12	12	6
	6	6-1	12	12	6	-	6	-	-	-	-
		6-2	-	-	-	12	6	6	12	12	6
Total			72	72	36	72	72	36	72	72	36
Above the core	1	1-1	12	6	6	6	-	-	-	-	-
		1-2	-	-	-	6	6	6	12	6	6
	4	4-1	12	6	6	6	-	-	-	-	-
		4-1	-	-	-	6	6	6	12	6	6
Total			24	12	12	24	12	12	24	12	12
Control set	1	1K	18	15	6	18	15	6	18	15	6
	2	2K	18	15	6	18	15	6	18	15	6
Total			36	30	12	36	30	12	36	30	12
In all			132	126	60	132	126	60	132	126	60

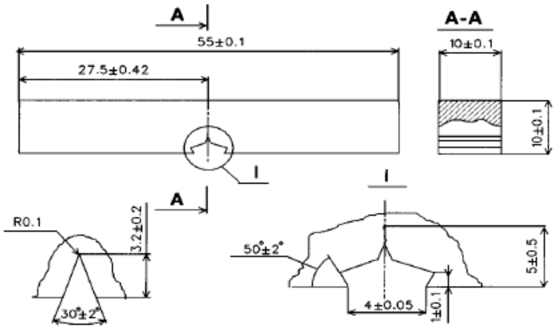
Geometry of the specimens



Tensile

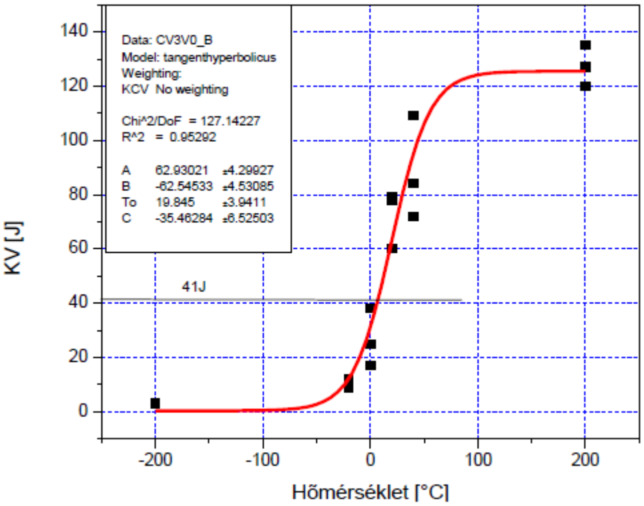


Charpy



Bending
(fracture mechanics)

Specimen examination (example)

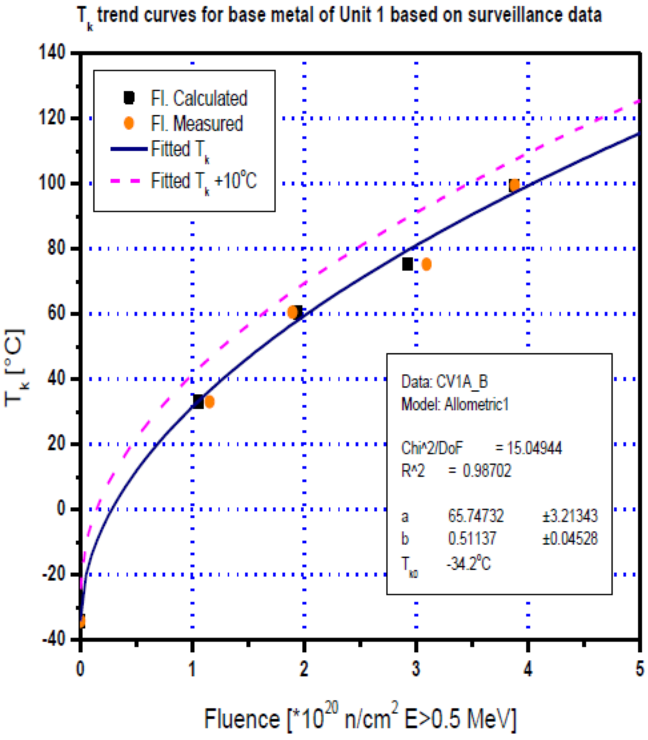


$$KV = A + B \cdot th\left[\frac{T - T_0}{C}\right] \quad , \quad [J]$$

A, B, C, T_0 = constants
T = examination temperature

Typical tangent hyperbolic fitting and marking of the criterion 41 J

Trend curve for base metal, Unit 1



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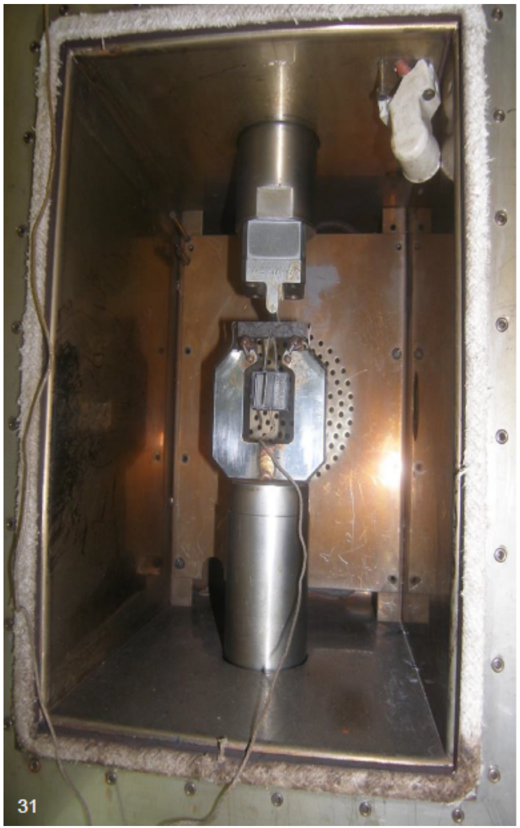

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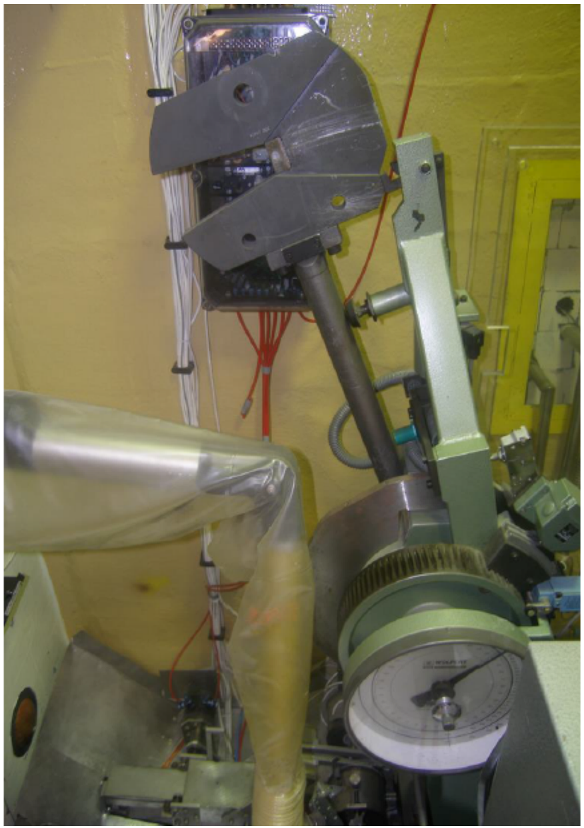

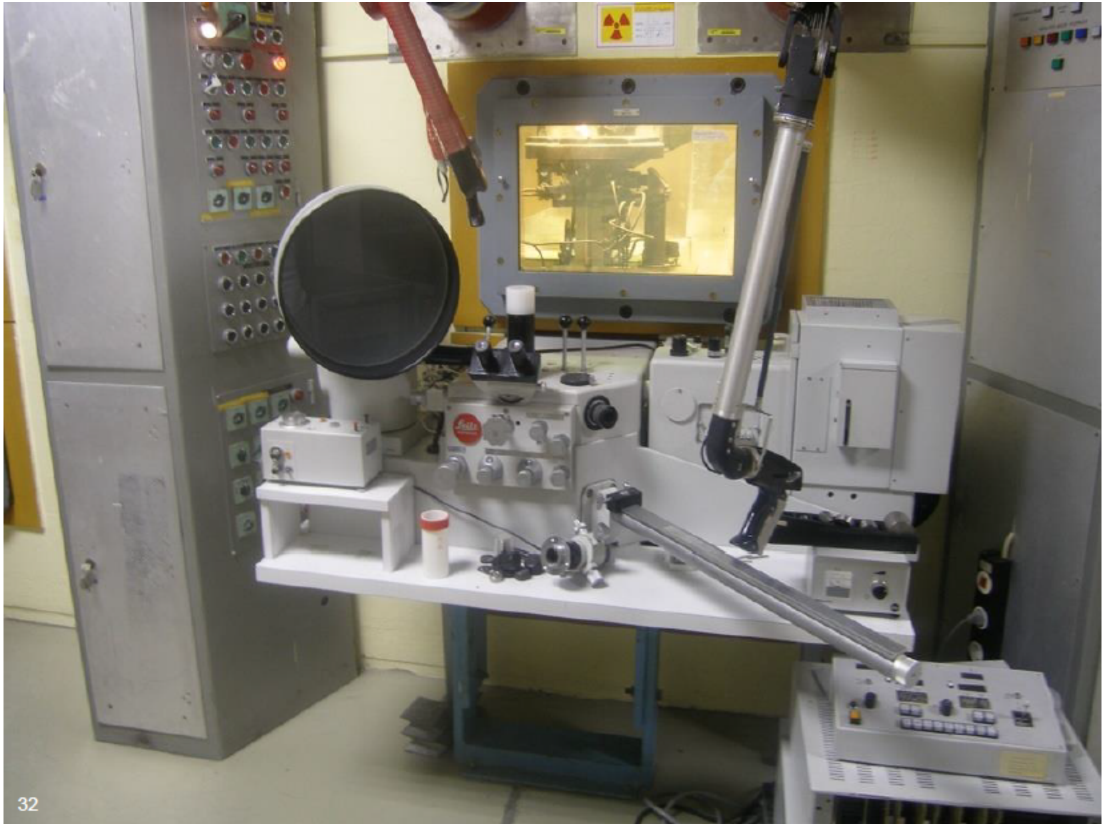

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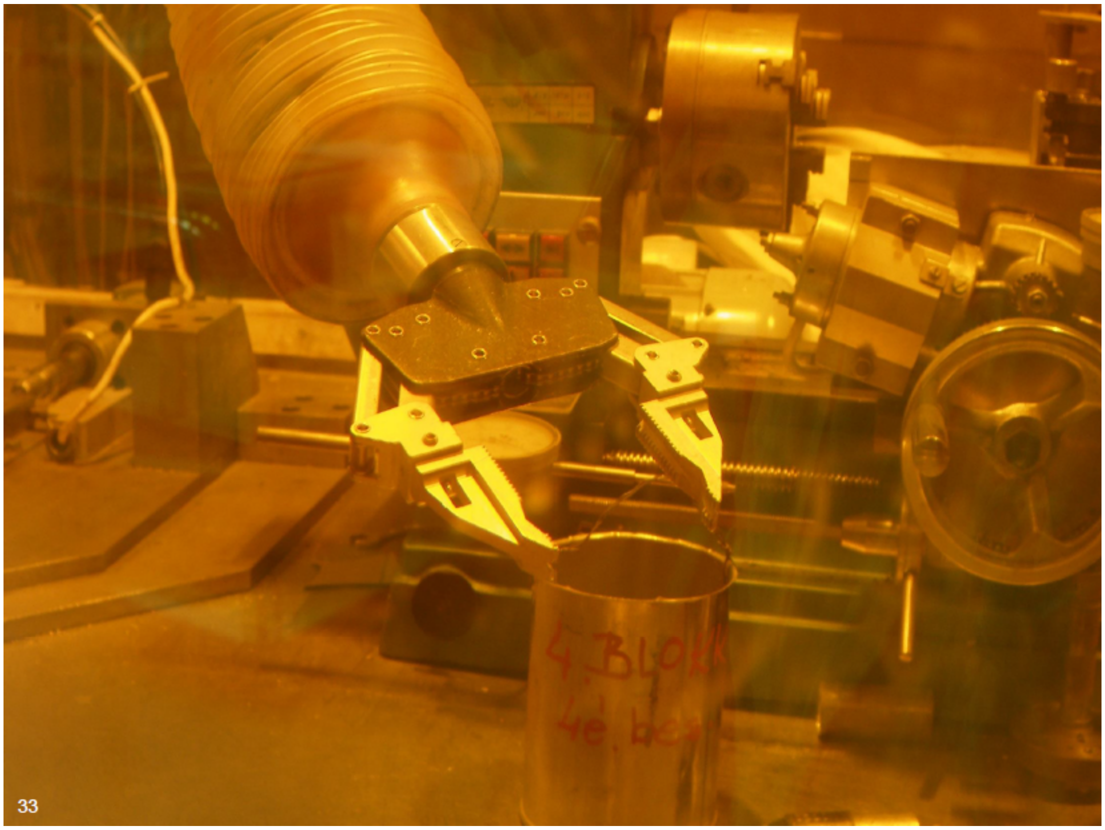
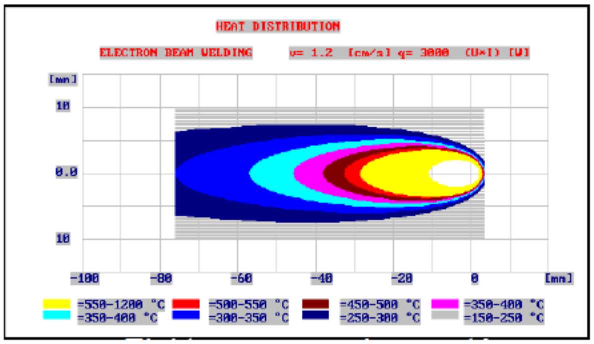
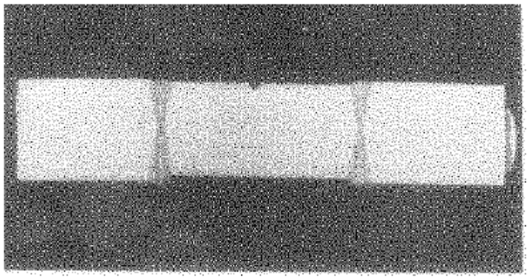


New surveillance programme No.1 (1991- 2012)

- Goal:
  - Monitoring the RPV condition after the original programme (for comparison)
  - Eliminate the gaps of the original programme
- Placing in the empty channels
- Materials:
  - Using original surveillance (15Ch2MFA), reconstituted impact specimens
  - Non-original RPV material (15Ch2MFA)
  - IAEA reference material (JRQ)
- Irradiation term: 4 (5) years

Reconstitution of Charpy impact specimens





New surveillance programme No. 2 (2013 - 2030)

- Goal:
  - Continuous monitoring the RPV condition for the extended life (50 years)
  - Results comparison to the original and NSP-1 programmes
  - Application of ASTM E-185-09a and ASTM E-2215-08 standards
- Materials:
  - Using original surveillance (15Ch2MFA), reconstituted impact specimens
  - Greifswald cladding specimen

New surveillance programme – 2 withdrawal plan

UNIT	CAMPAIN	LOADING	WITHDRAWAL	CHAIN ID.
1.	29-32.	2011	2015	4G1E, 4G2E
	29-36.	2011	2018	5G1E, 5G2E
	29-44.	2011	~2027	6G1E, 6G2E
2.	28-31.	2012	2016	4G1F, 4G2F
	28-35.	2012	2020	5G1F, 5G2F
	28-43.	2012	~2028	6G1F, 6G2F
3.	29-32.	2013	2018	4G1G, 4G2G
	29-36.	2013	2022	5G1G, 5G2G
	29-44.	2013	~2029	6G1G, 6G2G
4.	28-31.	2014	2017	4G1H, 4G2H
	28-35.	2014	2021	5G1H, 5G2H
	28-43.	2014	~2030	6G1H, 6G2H

Research programme for RPV internals: TLAA due to void swelling and IASSC

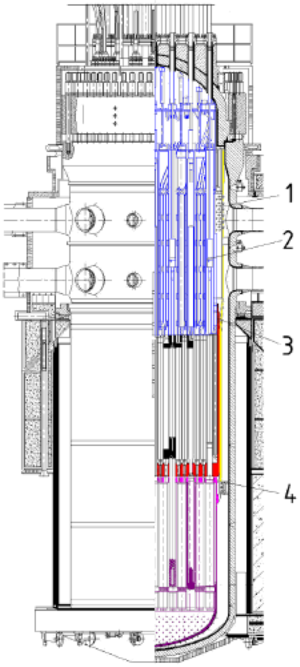
RPV Internals research for LTO (TLAA)

Degradation mechanisms *(due to irradiation)*	**Material, Locations**
▪ void swelling; ▪ IASCC; ▪ loss of fracture toughness; ▪ H_2 and He generation radiation induced segregation; ▪ stress relaxation (creep);	**Materials:** 08Ch18N10T (VVER stainless steel, similar to AISI 321) **Locations:** RPV internals: (core basket, baffle and former plates, baffle-to-former bolts) 

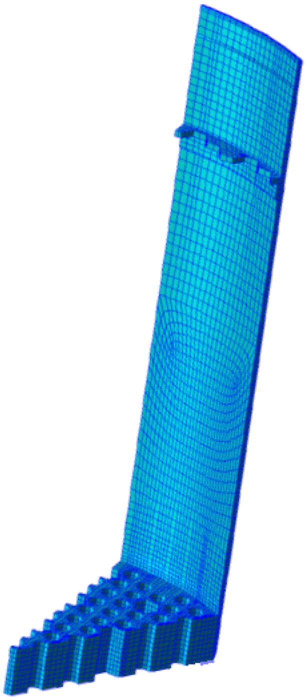
Reactor internal structures: New TLAA (1)

Current situation based on the previous TLAA:

- Local environmental influences (neutron and gamma irradiation, temperature) can lead to **void swelling degradation of reactor pressure vessel internal components**, which later may result in a local geometric change and deformation constrain
- In individual structural elements cannot be excluded the appearance of **stress corrosion cracks during the LTO period**
- For **irradiation-assisted stress corrosion, critical location** can be the baffle-to-former bolts of the basket
- Potential corrosion cracks in the bolts and deformation constrain due to the void swelling in extreme cases may **lead to the damage of thread part and loss of bolts.**



Reactor internal structures: New TLAA (2)



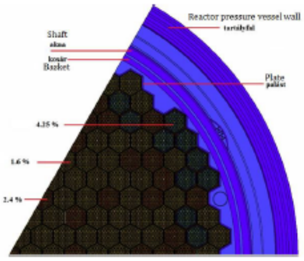
Goals of the new TLAA:

- Make a **more realistic estimation of the expected damage** with eliminating the unnecessary conservativism
- The **actual irradiation temperature of the baffle-to-former bolts, the baffle plate, and basket plate will be determined**, taking into account real gamma heating and local flow conditions. **It is assumed that knowing the real irradiation temperature can be significantly reduced the previously estimated void swelling level of the components.**
- Use more precisely **finite element calculation**

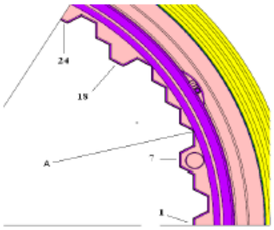
Reactor internal structures: New TLAA (3)

Main tasks to be performed:

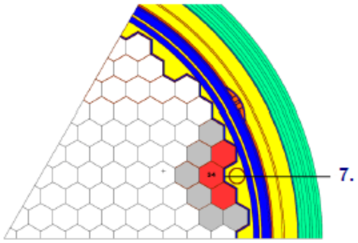
- **reactor - physics calculation:** 3D transport calculations for the fast neutron and gamma radiation;
  - irradiation load calculation → determine the radiation load in dpa to the internal elements of the reactor.
- **calculations for damage and gamma heating:** based on reactor-physics calculations (neutron flux and spectrum data) follow the neutron and gamma processes inside the construction materials;



The horizontal cross-section of the model



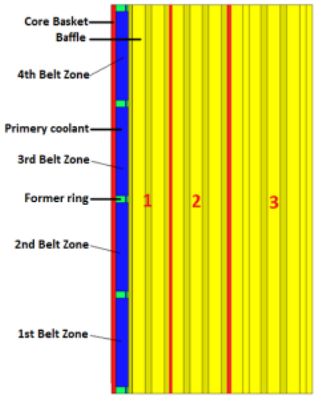
Numbering of sheets of baffle plate



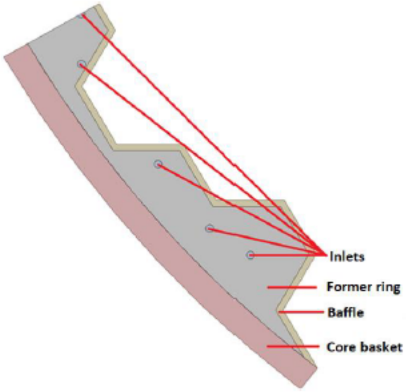
In red: cassettes providing ~95% of the heat generation on sheet 7.

Reactor internal structures: New TLAA (4)

- **CFD calculations:** heat and flow analysis to determine the characteristic irradiation temperature of the reactor vessel internals for the swelling analysis. The purpose of the model is to refine the irradiance temperature by calculating the boundary conditions, considering gamma heat sources.



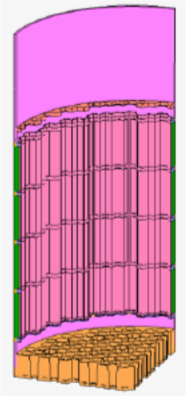
Side view of the examined area



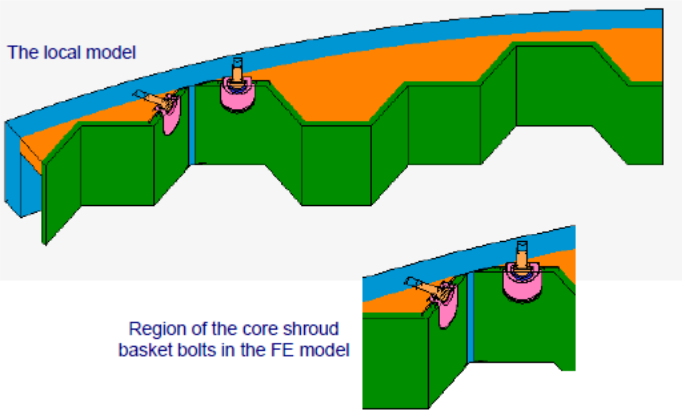
Geometric model in bottom view

Reactor internal structures: New TLAA (5)

- **FE modelling and swelling:** the FE models use at the life extension CUF calculation act as base models, and extend so that swelling could be taken into account within the Msc. Marc code. In addition the models are expand and specify in the regions expected high rate of swelling.



Global model of the basket



Region of the core shroud basket bolts in the FE model

Reactor internal structures: New TLAA (6)

- **Swelling calculation**: based on the available three swelling models

Classical VVER model $S_0 = c_D D^n exp[-r(T_{irr} - T_{max})^2]$,

D – radiation load [dpa],
T_{irr} – radiation temperature [°C],
$T_{max} = 470°C,$
$n = 1{,}88,$
$c_D = 1{,}035 \cdot 10^{-4},$
$r = 1{,}5 \cdot 10^{-4} °C^{-2}$.

Kalchenko model[1] $S_0 = (0{,}25 - 0{,}022lnk) \cdot \varphi(D - 103 + 0{,}1T - 2{,}6lnk) \cdot exp\left(-\frac{(T - 690 - 15{,}5lnk)^2}{2 \cdot (12{,}3 - 1{,}9lnk)^2}\right)$

D – radiation load [dpa],
T_{irr} – radiation temperature [°C],
k – radiation dose rate [dpa/s],
$\varphi(x) = x$, if $x > 0$, otherwise: 0.

---

[1] A.S. Kalchenko, V.V. Bryk, N.P. Lazarevó, I.M. Neklyudov, V.N. Voyevodin, F.A. Garner, Prediction of swelling of 18Cr10NiTi austenitic steel over a wide range of displacement rates, Journal of Nuclear Materials, 399 (2010), 114 – 121

Reactor internal structures: New TLAA (7)

- **Void swelling calculation**: based on the available three swelling models

VERLIFE model[2]

$S = S_0 f_1(\sigma_{eff}) \cdot f_2(æ_p)$, where

$S_0 = cF^{nv} exp[-r(T_{irr} - T_m)^2]$,
F – radiation load [dpa],
T_{irr} – radiation temperature [°C],
$T_m = 470°C,$
$nv = 1{,}88,$
$c = 2{,}588 \cdot 10^{-4},$
$r = 1{,}825 \cdot 10^{-4} °C^{-2};$

$f_1(\sigma_{eff}) = 1 + P \cdot \sigma_{eff},$
$\sigma_{eff} = (1 - \eta_1)\sigma_m + \eta_1 \cdot \sigma_{eq},$
$P = 5{,}4 \cdot 10^{-3}\ 1/MPa,$
$\eta_1 = 0{,}15,$
$\sigma_m = (\sigma_1 + \sigma_2 + \sigma_3)/3$ hydrostatic stress,
σ_{eq} – Mises's equivalent stress;

$f_2(æ_p) = \exp(-\eta_2 æ_p),$
$\eta_2 = 8{,}75.$

---

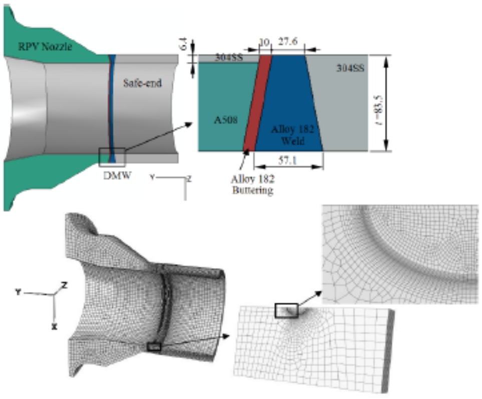
[2] Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plant (VERLIFE), Appendices A-F (2014)

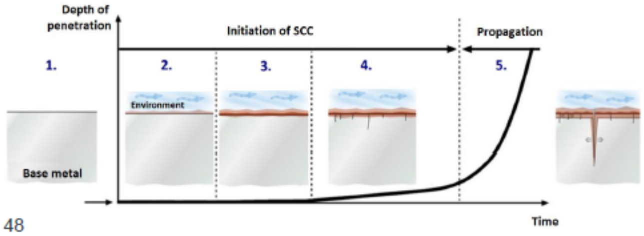
Other research programmes for Stress Corrosion Cracking of primary coolant pressure boundary

Prediction possibilities of stress corrosion crack initiation and its propagation

The project (2018-2021) focuses on the following topics:

- For primary circuit materials (dissimilar metal welds and baffle-to-former bolts due to high irradiation) study the new methods and results concerning
  - stress corrosion cracking (initiation) and crack propagation,
  - modelling and testing of SCC.
- Perform finite element analyses for the application of fracture models and for the determination of proposed parameters for analysis.





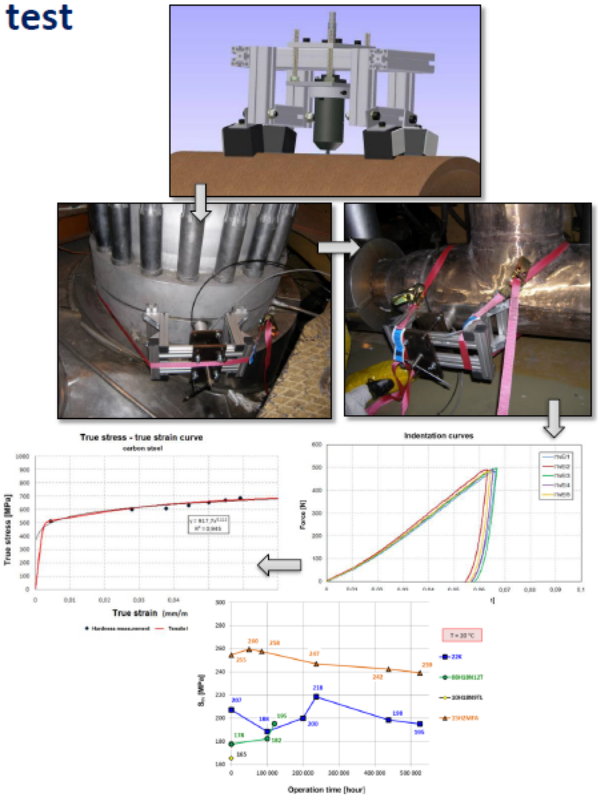
- Collect and determine by tests the specific material characteristics required for the applied model calculations.
- A method is developed for transferring test results to real size and complex geometry.

Research programme for verifying the material properties of the primary circuit materials

Instrumented indentation test of the primary pressure boundary components

For the most typical materials made of the Class 1 components (15Ch2MFA, 22K, 08Ch18N10T, 08Ch18N12T) to perform local instrumented indentation tests and evaluating the results in order:

- to control the **material properties during LTO period** (yield strength and tensile strength),
- to compare the **strength values with the applied ones in the current ASME BPVC III based calculations** (complete re-design was performed in advance the plant entered into LTO period).

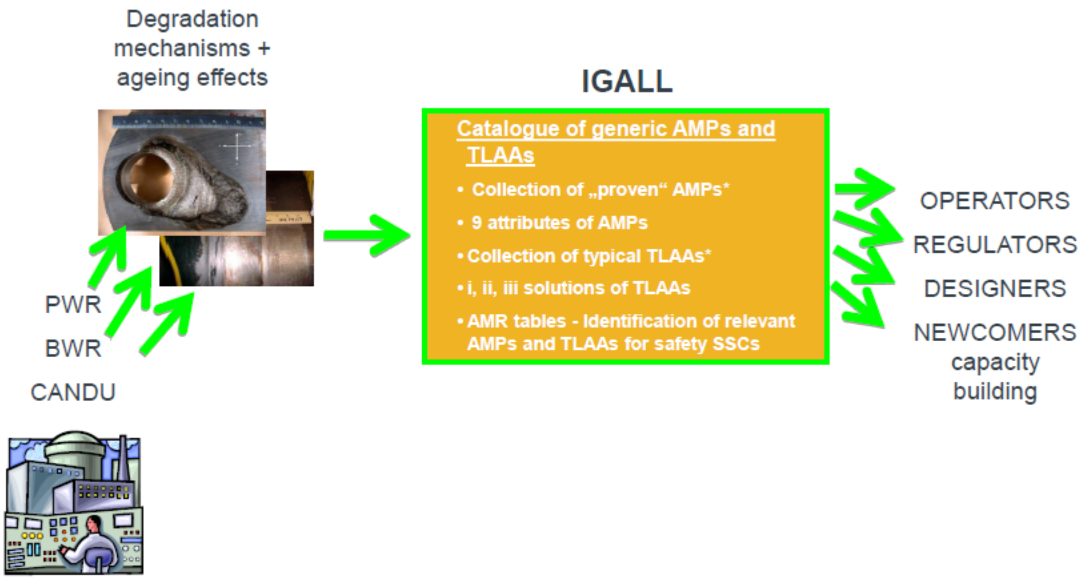


Other R&D activities: cooperation with EPRI, IAEA and European Union

EPRI cooperation

- Extension of EPRI Material Degradation Matrix (MDM) with VVER materials;
- Expert opinion of RPV internals TLAA methodology and criteria document (later results)
- Examination of steel samples exposed to high radiation
- *Outside the primary pressure boundary locations: Adaption and application of the EPRI Erosion Corrosion programme (CHEKWORKS)*

Participation in the IAEA IGALL programme



Degradation mechanisms + ageing effects

PWR

BWR

CANDU

IGALL

Catalogue of generic AMPs and TLAAs

- **Collection of „proven“ AMPs***
- **9 attributes of AMPs**
- **Collection of typical TLAAs***
- **i, ii, iii solutions of TLAAs**
- **AMR tables - Identification of relevant AMPs and TLAAs for safety SSCs**

OPERATORS

REGULATORS

DESIGNERS

NEWCOMERS capacity building

Robert Krivanek, IAEA , LTO programme officer

R&D cooperation with EU

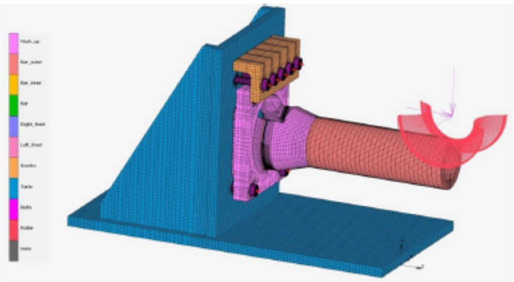
- NUGENIA (**NU**clear **GEN**eration II & III **A**ssociation) *(International non-profit association)*
- NUGENIA Hungarian members are MVM Paks Nuclear Power Plant Ltd., Bay Applied Research Ltd., MTA EK. The organization includes and carries out SNETP, NULIFE and SARNET programmes and their results.
- The results of R&D programmes contribute to work out the technical-scientific basis of the Paks NPP AMPs.

Example: ATLAS programme

ATLAS: Advanced Structural Integrity Assessment Tools for Safe Long Term Operation

Hungarian contribution:

- Simulation-based design of experiments on medium-sized mock-up
- Deeper understanding of damage phenomena. Better understanding of the transferability issues, real constraint conditions, WRS effect on crack initiation and ductile tearing, especially under loading from accident conditions.
- Analysis of main coolant pipe nozzle of RPV (VVER-440)

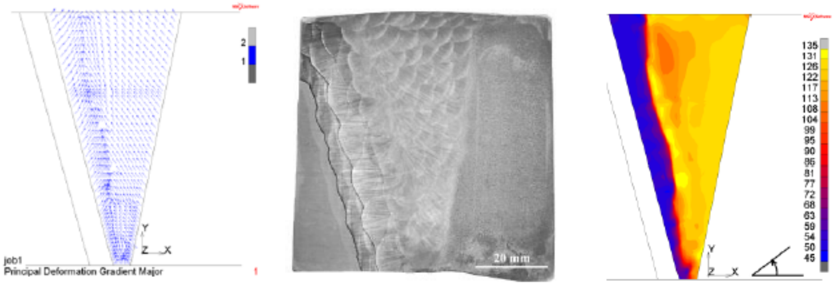


Example: ADVISE programme

ADVISE - Advanced Inspection of Complex Structures. ADVISE aims to advance the ultrasonic inspection of complex structured materials, for which conventional ultrasonic techniques suffer from severe performance limitations due to the micro and/or macro-structure

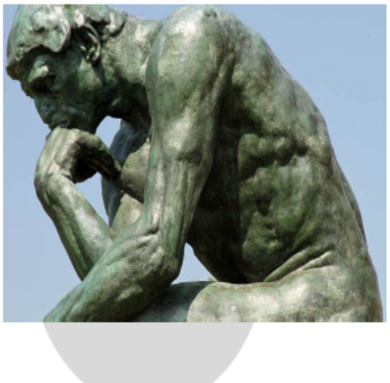
Hungarian contribution:

- Develop 3D simulation models to predict the weld formation, including maps of stiffness orientation and grain size; provide maps in parametric form and simplified numerical 3D method that will be suitable for inversion methods
- to integrate the components developed in the preceding WPs to carry out the transfer from the laboratories involved to the actual application in the field



Summary

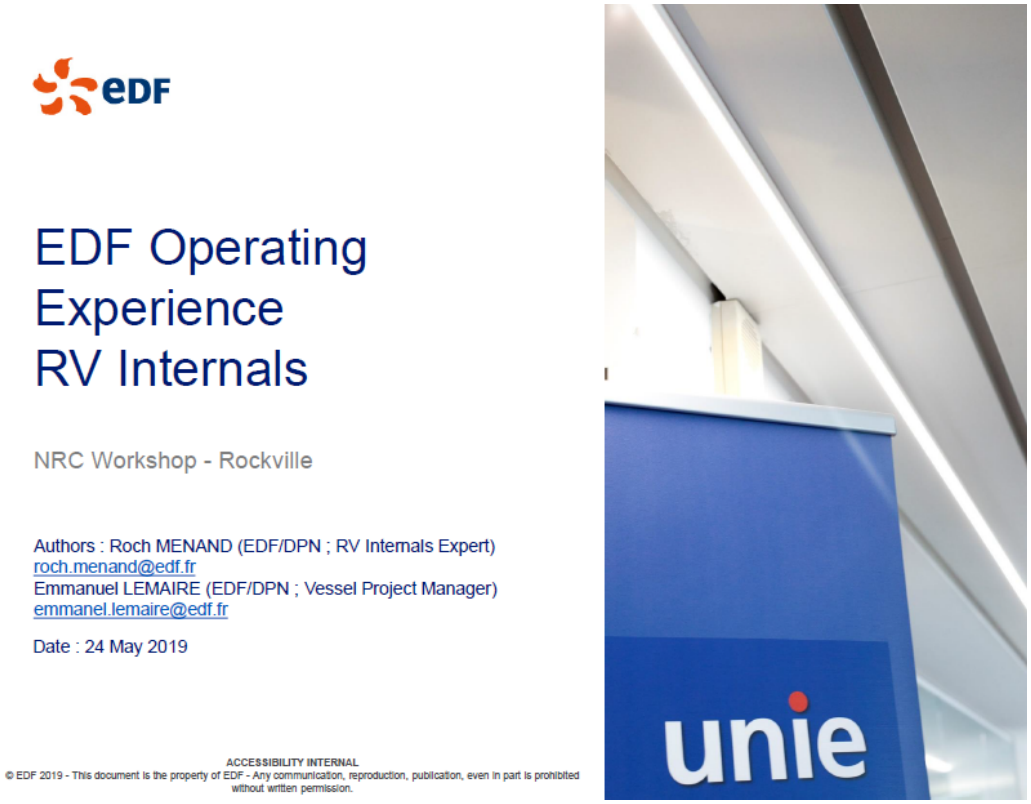
- Technical basis of LTO at MVM Paks NPP is established:
  - ➢ Ageing management
  - ➢ Time Limited Aging Analyses (TLAAs)
- The plant has gained the extended licenses for the next 20 years
- Research needs are identified for the safe LTO
- Hungary joined to a couple of international programmes in the field of ageing management (e.g. EPRI, IAEA, EU)



English: Rodin's *The Thinker* at the Musée Rodin
Français: *Le Penseur* dans le Musée Rodin.

Thank you for your attention!

B.21 EDF Operating Experience RV Internals (R. Menand)



EDF Operating Experience RV Internals

NRC Workshop - Rockville

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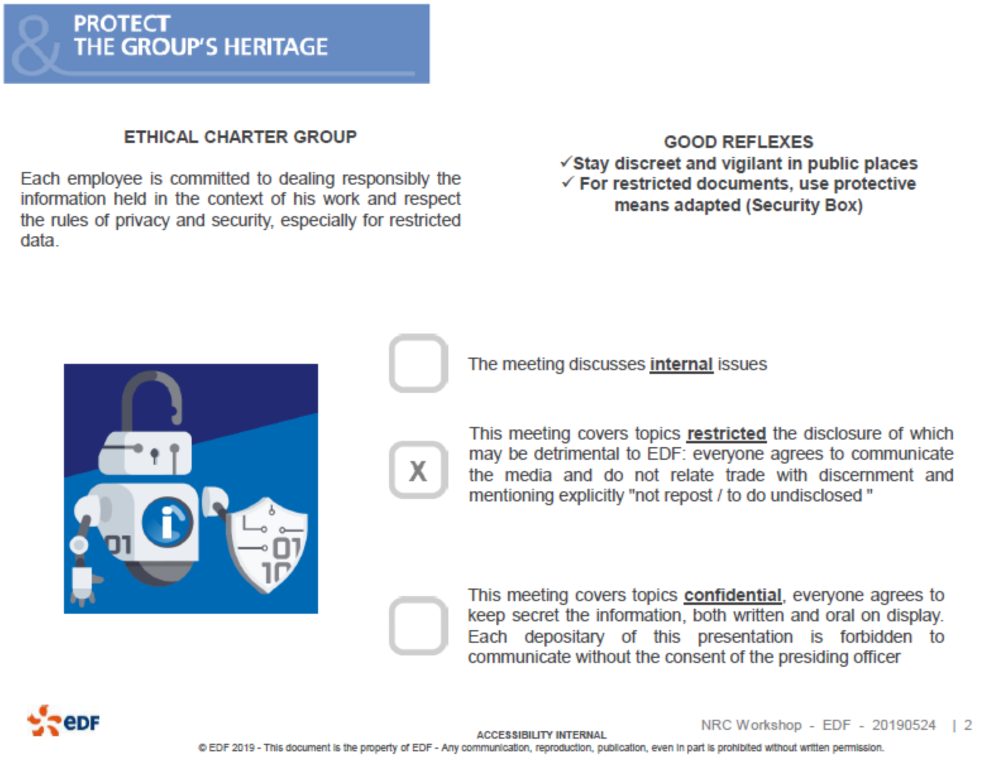
Date : 24 May 2019

ACCESSIBILITY INTERNAL



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- ✓ **Stay discreet and vigilant in public places**
- ✓ **For restricted documents, use protective means adapted (Security Box)**

☐ The meeting discusses **internal** issues

☒ This meeting covers topics **restricted** the disclosure of which may be detrimental to EDF: everyone agrees to communicate the media and do not relate trade with discernment and mentioning explicitly "not repost / to do undisclosed "

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ACCESSIBILITY INTERNAL

SUMMARY

ACCESSIBILITY INTERNAL

EDF'S NUCLEAR PRODUCTION IN FRANCE

In operation NPP's	Number of units	Total capacity (GWe)	Average age (years)
PWR 900 MWe 3-Loop	34	30.8	33
PWR 1300 MWe 4-Loop	20	26.4	30
PWR 1450 MWe 4-Loop (N4)	4	6.0	20
Total	58	63,2	

- **58 reactors in operation**
  - ✓ 19 sites
  - ✓ Capacity : 63,2 GWe
  - ✓ Average age : 30 years
- **✓+ 1 reactor under construction**
  - ✓ FLA 3, 1600 MWe EPR

■ 900 MWe PWR series
△ 1300 MWe PWR series
▽ 1450 MWe PWR series
▲ 1600 MWe PWR series (under construction)

ACCESSIBILITY INTERNAL

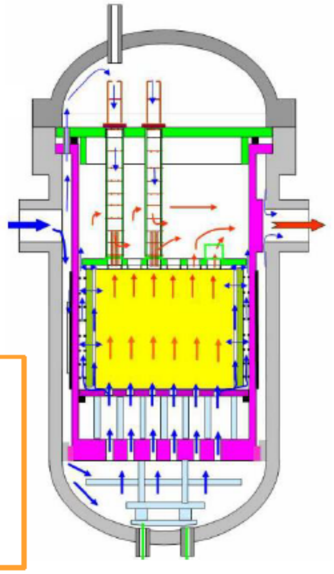
REACTOR INTERNALS FUNCTIONS

Reactor internals functions:

- Provide support, guidance and protection of the rod control cluster assemblies (RCCA).
- Provide support and orientation to the reactor core (fuel assemblies).
- Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
- Provide a passageway for support, guidance and protection for in-vessel/core instrumentation.

Reactor Safety functions:

- Reactivity control
- Core cooling control
- Radioactive material containment control



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REACTOR INTERNALS FUNCTIONS

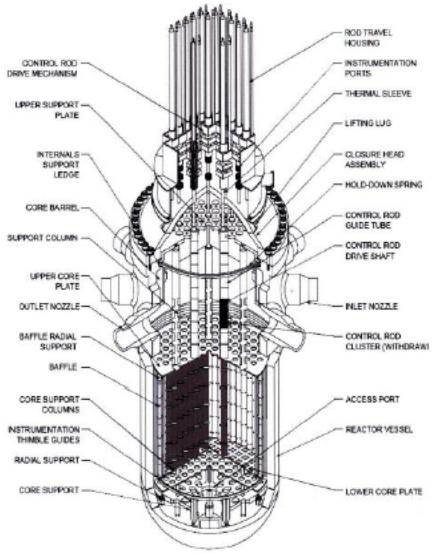
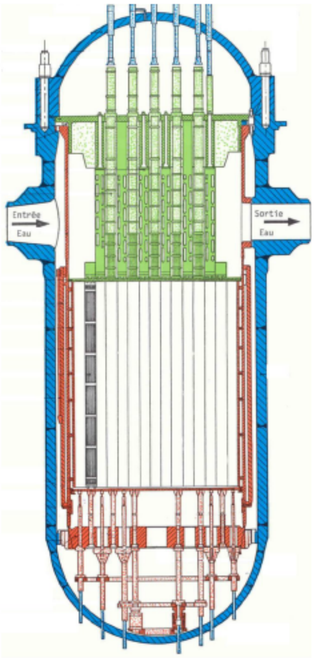


Figure 3-5
Overview of typical Westinghouse internals

3-loop RV Internals :

- Material Temperature : 290 to 355°C
- Upper Internals : 45 tons
- Lower Internals : 90 tons
- Height of Fuel assemblies : 4076 mm
- # Fuel Assemblies : 157
- Core barrel diameter : 3400 mm



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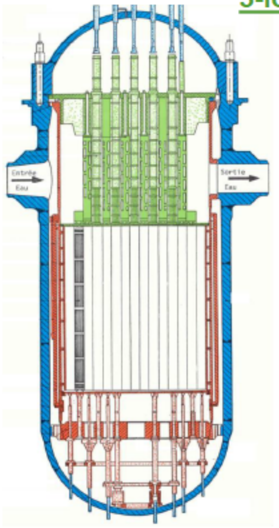
REACTOR INTERNALS FUNCTIONS

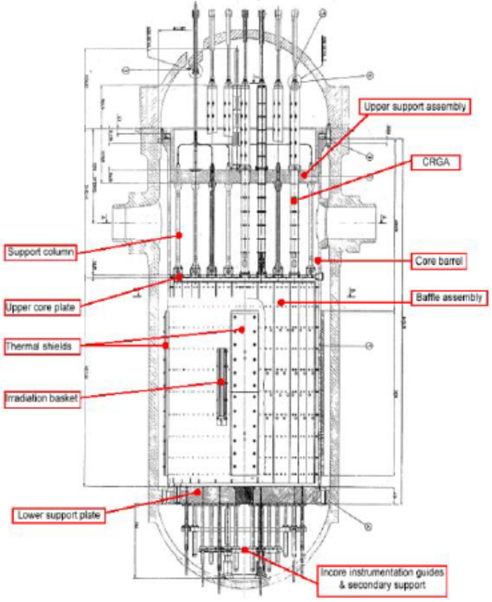
3 loops



3-loop vs 4-loop PWR :

- Temperature into the vessel head plenum (to avoid SCC)
- Upper support assembly plate (less fatigue)
- Baffle assembly (more fuel assemblies)
- Thermal shields (thickness of the water)
- Lower Core Plate (height of the fuel assemblies)

4 loops



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REACTOR INTERNALS FUNCTIONS

- **Thin structures (no pression boundary)**
- **Materials : Stainless Steel (304 or 316 Cold Worked)**
- **Vibrations => every pieces must be « screwed or welded »**
- **High temperatures (nearly 350°C – gamma heat)**
- **High Irradiations (some components are very close to the fuel assemblies)**

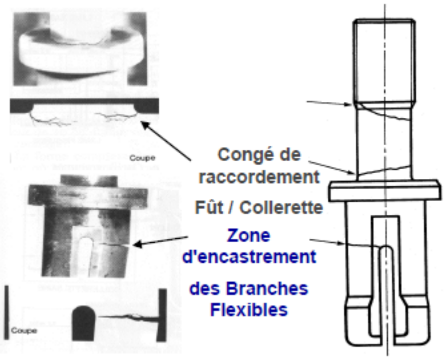
ACCESSIBILITY INTERNAL

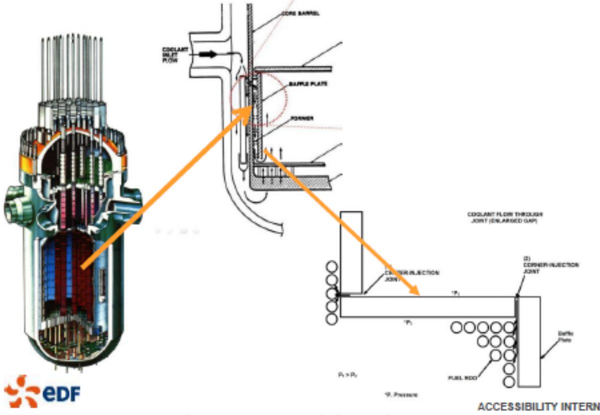
REACTOR INTERNALS FUNCTIONS

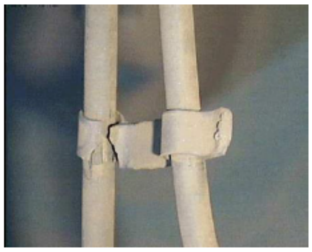
- **The RVI components must fulfil an Ageing Management Program (AMP) in agreement with the French regulation for the NPP third decennial outages and the following decennial outages. As for other Systems and Components, the AMP is carried out at EDF in 3 main steps, which are globally in agreement with IAEA Safety guide NS-G-2.12 on Ageing Management for Nuclear plants :**
  - Selection of Safety Structures, concerned by an ageing mechanism (SSC),
  - Review of all the couples SSC / degradation mechanisms selected by experts and run synthetic analyses on Ageing Analysis Sheets, taking into account maintenance adaptability, difficulty to repair or replace as well as risks of obsolescence,
  - Detailed Ageing Management Reports in order to justify that all the components concerned by an ageing mechanism can operate within the safety criteria during the considered period of operation (10 years).
- **Because they don't contribute to pressure resistance, installed RV Internals are not submitted to French regulation on pressurized equipment.**

STATE OF KNOWLEDGE - OE

- Stress Corrosion Cracking (SCC) :
  - Baffle jetting (1980) => upflow conversion
  - SCC in the split pins of the guide tubes (1982, Inconel X750)
    => replacement with new pins design since 1990s
  - SCC in the thermocouples clamps (2008, SS 316)
    => extraction of clamps to avoid loose part

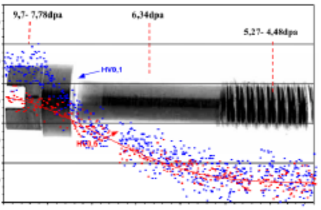


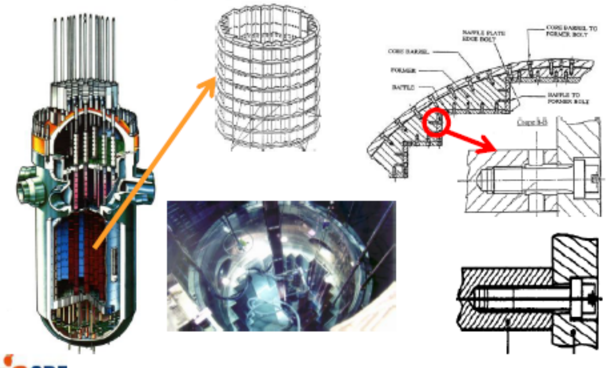




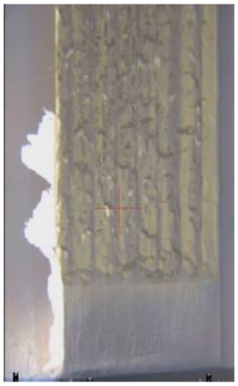
STATE OF KNOWLEDGE - OE

- **Effect of Irradiation :**
  - IASCC on the Baffle Former Bolts (1990s -> today)
    => temperature and irradiation modelling
    => examination on many bolts (10 to 40 dpa)
  - Potential swelling on baffles and bolts
    => ongoing international research (GONDOLE program, R&D modelling…)
  - Loss of ductility, segregation, relaxation…

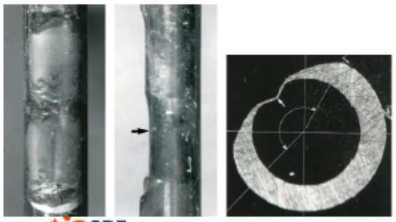


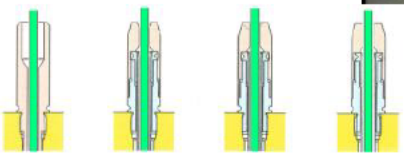


STATE OF KNOWLEDGE - OE

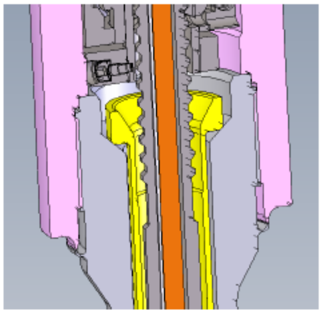


- **Wear of several components :**
  - Wear and leak of the flux thimble tubes (1985-89)
    => raising of the thickness of FTT, lowering the gap between FTT and the lower core plate
  - Wear of some lower core pins (1999)
    => happened on one single plant, no real issue
  - Wear of the stellite surfaces of radial keys (2010)
    => 3D calculations to increase the admissible gap in case of accident (LOCA and seismic conditions)
    => VT-1 on each decennial outage + some wear measurement

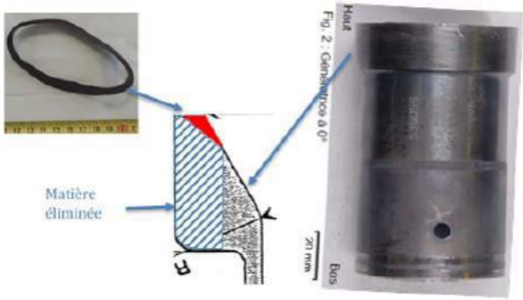




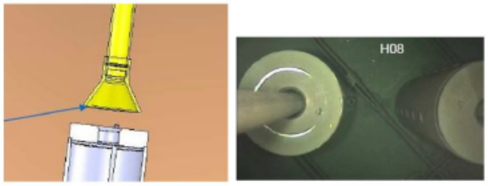
STATE OF KNOWLEDGE - OE



- **Wear of several components :**
  - Wear of guide tubes (2012)
    => wear measurement on 4 central canals
    => replacement of guide tubes
  - Wear of thermal sleeve of the vessel head (2017)
    => lowering measurement under every RV heads
    => replacement of the TS to avoid loose part
    => considering mitigation to stop TS wear and head penetration wear also



OE – THERMAL SLEEVE WEAR



- First quarter 2019, first wear measurement have been performed on each 58 French reactors (3L and 4L)
- Main results :
  - 3L-900 MWe CP0 (T-hot head) : No wear measured
  - 3L-900 MWe CPY (T-cold head) : No significant wear, moderate wear on some TS located in central positions (lowering < 30 mm after 22 years)
  - 4L-1300 MWe (T-cold head) : Significant wear, maximum in H08 (central position) but also significant on TS located near the center or in the periphery including non controlled positions.
  - 4L- 1450 MWe N4 (T-cold head) : No significant wear, moderate wear on one plant, lowering < 30 mm after 22 years
- Second quarter 2019, second wear measurement to come on some 4L-1300 MWe plants

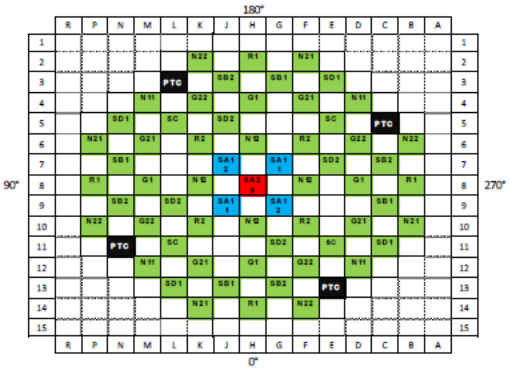
OE – THERMAL SLEEVE WEAR

Main results on the lowering measurement of more than 3300 thermal sleeve

OE – THERMAL SLEEVE WEAR

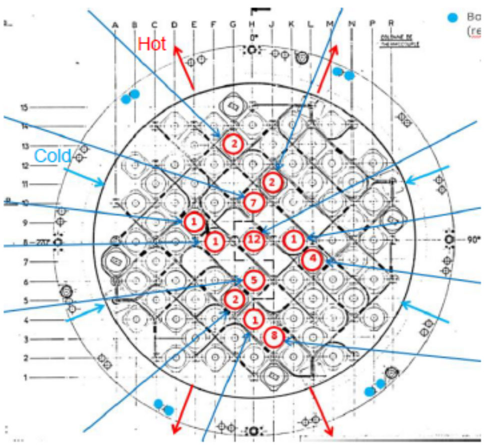
- **Location on the RV heads, difference between 3L and 4L Plants**

3-Loop



	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1																1
2						N22		R1		N21						2
3					PTC		SB2		SB1		SD1					3
4				N11		G22		G1		G21		N11				4
5			SD1		SC		SD2				SC		PTC			5
6		N21		G21		R2		N12		R2		G22		N22		6
7			SB1				SA1 2		SA1 1		SD2		SB2			7
8		R1		G1		N12		SA2 9		N12		G1		R1		8
9			SB2		SD2		SA1 1		SA1 2				SB1			9
10		N22		G22		R2		N12		R2		G21		N21		10
11			PTC		SC				SD2		SC		SD1			11
12				N11		G21		G1		G22		N11				12
13					SD1		SB1		SB2		PTC					13
14						N21		R1		N22						14
15																15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

4-Loop



OE – THERMAL SLEEVE WEAR

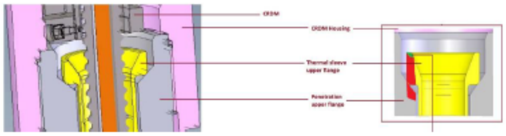
➢Immediate replacements performed during these inspections:

- 2017/12 initial event 4L-1300 MWe BEL2 : 15.5 EFPY (20 years calendar age)
- 2018 - 4L-1300 MWe NOG1 15.3 EFPY - Same issue as in Belleville, flange completely worn, control rod jammed, presence of the metal ring, the TS was supported by the GT.
- 2018 - 4L-1300 MWe SAL1 (17.3 EFPY) - marks on the top of the GT but flange not completely worn
- 2018 - 4L-1300 MWe PAL3 (14.4 EFPY) - marks on the top of the GT but flange not completely worn
- 2018 - 4L-1300 MWe BEL 1 (2000 – 14,2 EFPY) - significant wear but no loose part/complete wear
- 2019 - 4L-1300 MWe PAL 4 (1996 – 17,4 EFPY) – significant wear but no loose part/complete wear
- 2019 - 4L-1300 MWe FLA 2 (1998 – 15,7 EFPY) - significant wear but no loose part/complete wear

➢All the TS replacements have been achieved using a complete CRDM removal

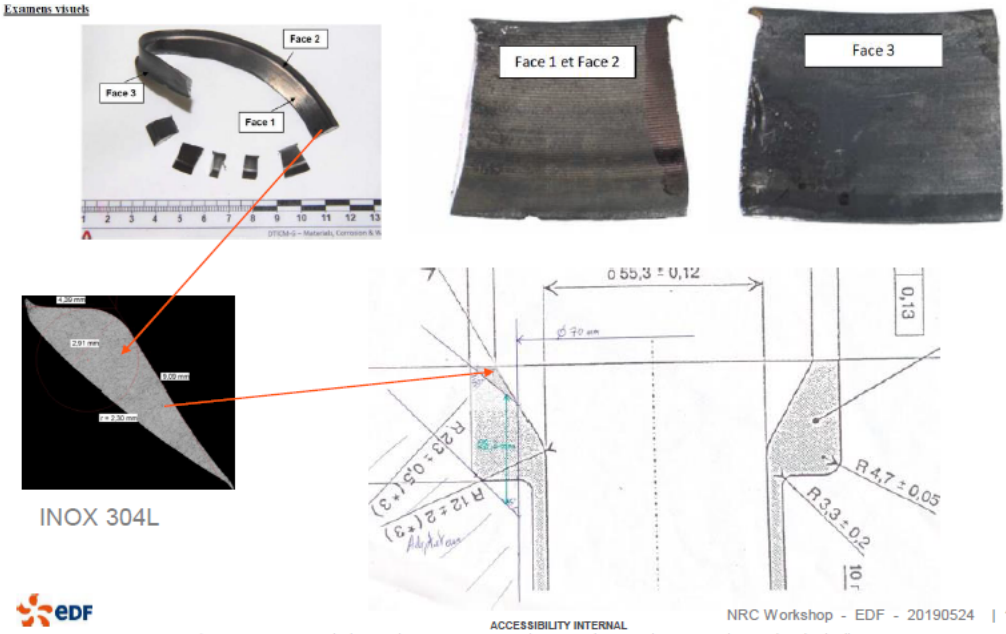
➢EDF is still working on the best way to optimize its maintenance and inspection strategy :

- A mechanical criterion is taken into account (39 mm max for TS lowering). On going studies for justifying penetrations without TS in locations without rods.
- The wear rate for extrapolated values is based on each RPV head and TS location specific measurements.



OE – THERMAL SLEEVE WEAR

- **Expertise of the loose part (ring), from Belleville 2 thermal sleeve**



INOX 304L

OE – THERMAL SLEEVE WEAR

- **Expertise of the thermal sleeve and the head penetration**

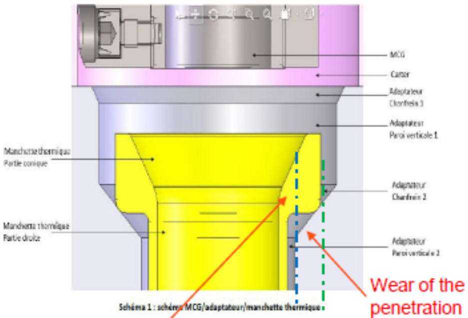


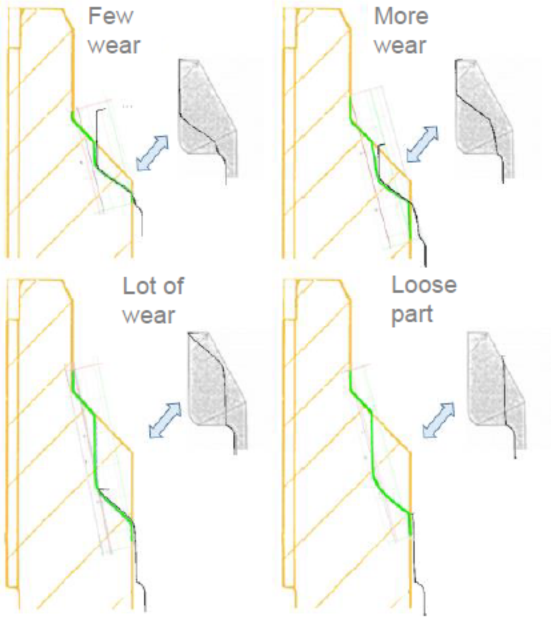
Schéma 1 : schéma MCG/adaptateur/manchette thermique

Wear of the penetration

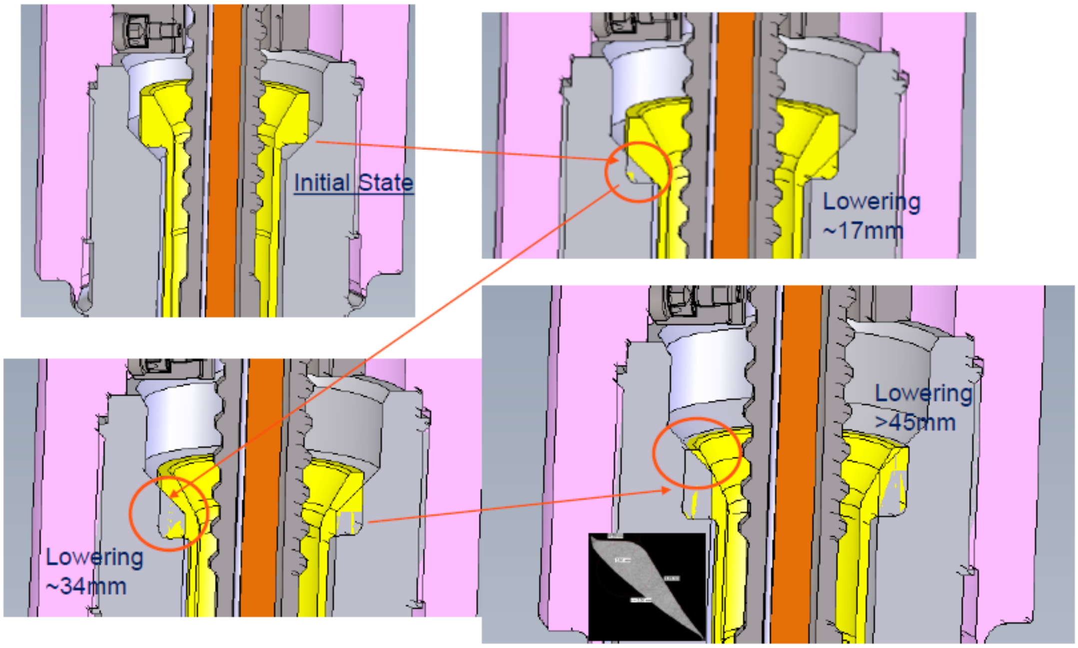
Wear of the TS

OE – THERMAL SLEEVE WEAR

- **The expertise of 17 thermal sleeves and head penetrations led to the reconstruction and knowledge of the different phases of wear (lowering between 20 to 50mm)**
  - The wear surfaces of the thermal sleeves have globally the same characteristics, on each lowering
  - Thin circular traces on all wear surfaces, smooth and without noticeable scaling => an **abrasion wear mechanism** (producing an effect similar to that of a machining) between the thermal sleeve and the head penetration under the effect of a rotating and/or ball-bearing movement of the thermal sleeve.



OE – THERMAL SLEEVE WEAR

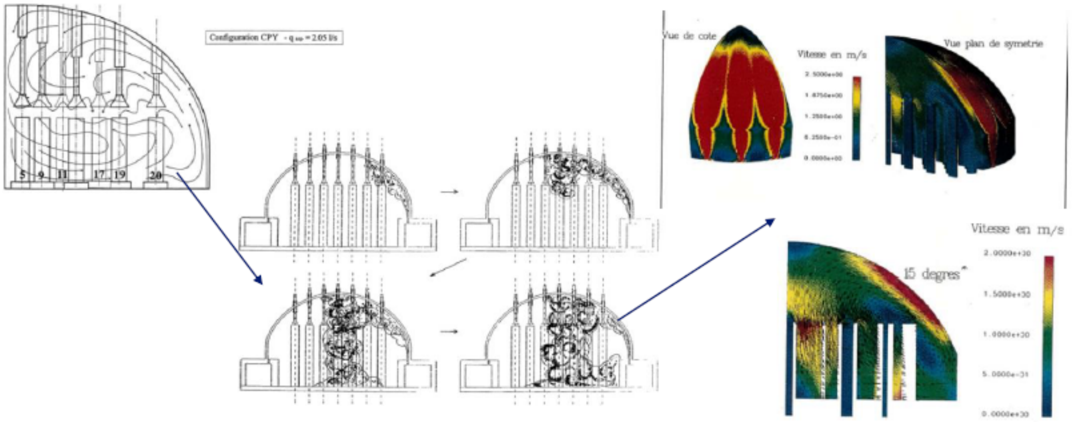


ONGOING RESEARCH – RV INTERNALS

- **RV Internals components can be classified in 3 types :**
- **Components with well known degradation, from a long time :**
  - Split pins of the guide tube, thermocouple clamps : SCC
  - Flux Thimble Tubes : wear
  - Guide tube : wear
- **Components with degradation from a long time, needing more knowledge :**
  - Bolts and baffles : IASCC, swelling and other irradiation material degradation
- **Components with rather “new” degradation, needing more knowledge :**
  - Radial key / clevis : wear on stellite surfaces
  - Thermal sleeve : wear on SS 304, thermohydraulic under the vessel head

ONGOING RESEARCH – THERMAL SLEEVE

- **Modelling and mechanical testing to understand :**
  - ☐ The initiation of the rotating and/or ball-bearing movement of the sleeve
  - ☐ The wear mechanism and the kinetics during the different steps
  - ☐ The location of the thermal sleeves worn in the RV head



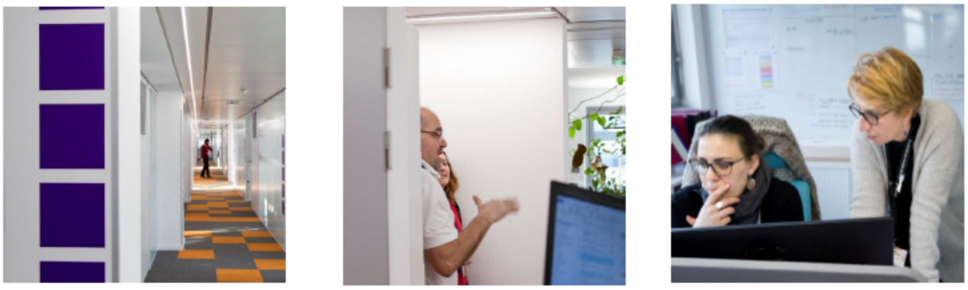
ONGOING RESEARCH – IRRADIATION

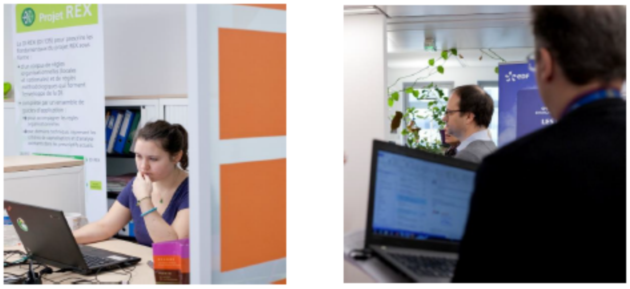
- **Objectives**
  - Loadings: determine the neutronic and thermal gradients on the Internal structures.
  - Material properties: advance in our understanding of material degradation mechanisms to better predict ageing kinetics.

- **Approach**
  - Loadings
    - ✓ Neutronics and thermo-hydraulics calculations to determine doses and temperatures of the components.
  - IASCC
    - ✓ Improve understanding of IASCC through both experiments and microstructure analysis (decommissioned components or experimentally irradiated samples).
  - Swelling
    - ✓ Determine PWR conditions leading to void swelling through simulations and observations.
    - ✓ Understand swelling mechanisms due to neutron irradiation through microstructure investigations.

LTO - CONCLUSION

- Improved safety and competitiveness of the NPP fleet are the main objectives of EDF lifetime policy in the frame of lifetime extension from 40 to 60 years.
- In this context, a detailed and systematic ageing management program (AMP) has been developed to review the ageing consequences on SSCs important to safety, applied since the 3rd 10-year safety review of 3L and 4L reactors.
- An effective maintenance policy (routine and exceptional) associated with qualified in-service inspections has been implemented, based on integrated feedback experience.
- Exceptional maintenance operations, such as guide tubes, split pins, CRDM and baffle bolts, are an illustration of EDF lifetime management policy, designed to prepare major industrial investment in the plant life extension context.





Thank you

B.22 Carbon Segregations in Heavy Forged Components (E. Viard)

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Enhancing nuclear safety

Carbon segregations in heavy forged components

IRSN assessment of non-conformances
in French PWRs primary coolant system

Emmanuel VIARD, IRSN/PSN/SES

NRC International workshop
May 23-24, 2019

MEMBER OF
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NETWORK

Content

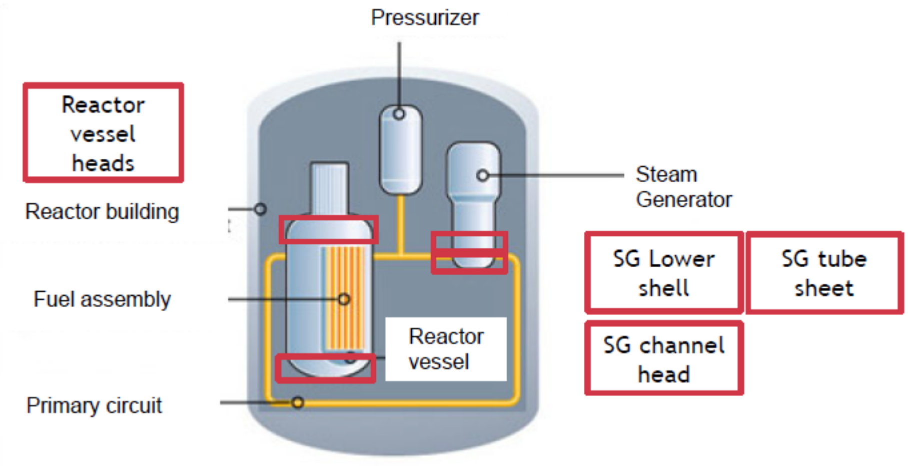
- Context and safety issues
- Manufacturing conditions
- Justification file
- IRSN position

Content

- Context and safety issues
- Manufacturing conditions
- Justification file
- IRSN position

Context

Concerned components in France



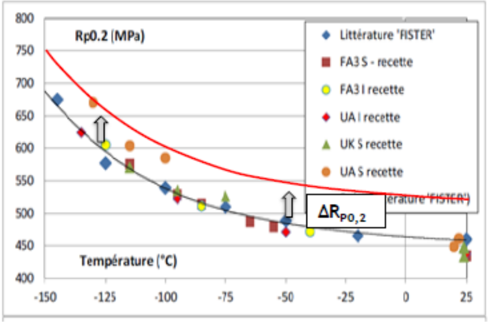
Context

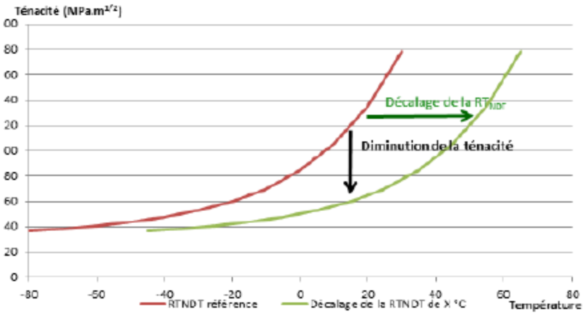
- Steel grade equivalent to A508
- RCC-M (French conception & manufacturing code) allows carbon content up to 0.25 % for this grade
- Carbon contents exceeding 0.3 % have been locally measured on some components during the technical qualification of heavy forged components (a French regulation requirement)
- Carbon contents exceeding maximum allowable values have been identified as the leading cause for low toughness values

↗ **A 508 mechanical properties considered in the code for design are questioned for these areas**

Safety issues

- A higher carbon content in A508 steels leads to :
  - An increase of tensile properties
  - A decrease of toughness properties => increase in DBTT





↗ **What are the properties of this material ?**

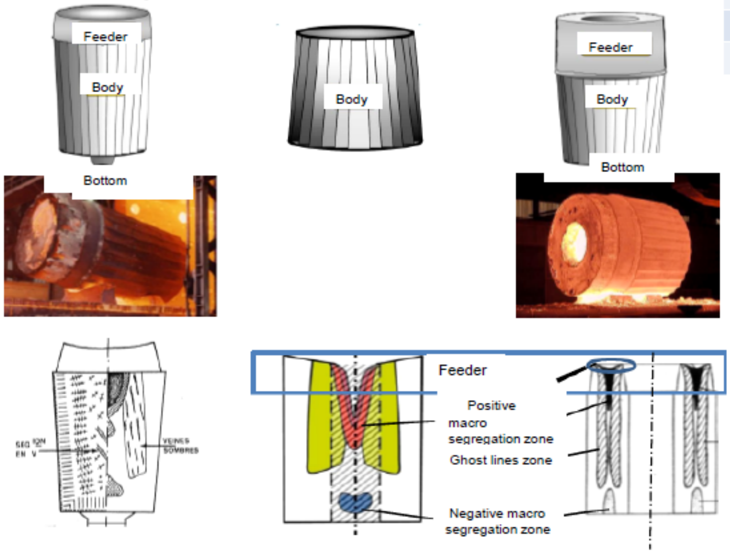
Content

- Context and safety issues
- Manufacturing conditions
- Justification data
- IRSN position

Manufacturing

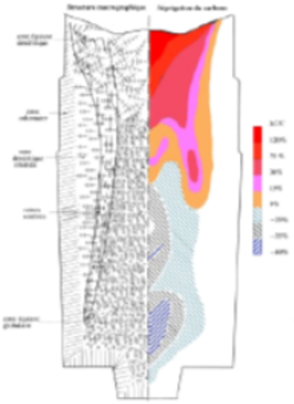
- Full ingot vs. Hollow ingot

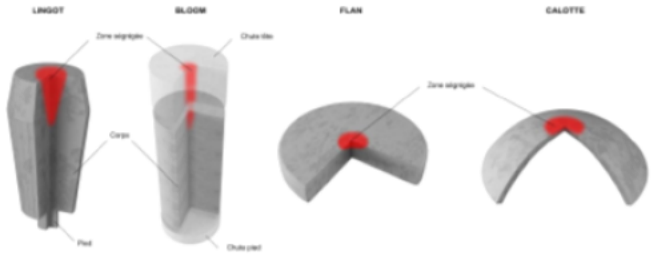
	Number of components
FA3 vessel heads	Full
SG channel heads	Full
SG lower shell	Hollow
SG tube sheets	Full



Manufacturing

- Full ingot - final localization of carbon segregation





Further manufacturing steps

- Manufacturing parameters have a major effect on final maximum carbon content
  - Ingot type and geometry (H/D ratio)
  - Feeder discard ratio
  - Machining
- Full ingot : evolutions of ingot type in the 90's led to larger and deeper segregation zone not sufficiently limited by subsequent manufacturing steps
- Hollow ingot :
  - Mistakes in feeder discard ratio and consequent machining (only 1 case in France i.e. SG lower shell)

Content

- Context and safety issues
- Manufacturing conditions
- Justification file
- IRSN position

Justification file

- French safety authority demanded to determine the toughness of the segregated areas based on sacrificial components characterization

- Reactor vessel heads : 3 components

- SG lower shell : 2 components

- SG primary heads
  - Justifications issued by EDF and accepted by French safety authority
  - Sacrificial components under characterization

- SG tube sheets
  - Justifications issued by EDF

Justification file

- Type of investigations
  - Carbon content mapping to identify the extension and intensity of the segregated area
  - Tensile tests
  - Toughness tests
    - Charpy V notch samples
    - CT samples

- Objectives :
  - Determine the potential shift in mechanical properties
  - Determine the risk of cold cracking in case segregated material is located in the vicinity of welds (case of SG lower shell)

Results - RV heads

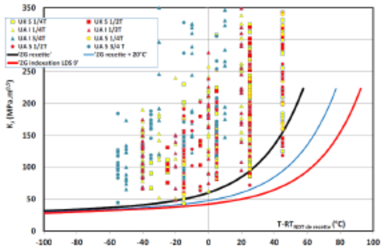
- Segregated area extent

	Component #1		Component #2			Component #3		
Zone	¼ T	½ T	¼ T	½ T	¾ T	¼ T	½ T	¾ T
C% (avg)	0,25	0,22	0,27	0,25	0,22	0,28	0,27	0,23

- Mechanical properties

	ΔRp0,2 (Mpa)	ΔRm (Mpa)	A %	ΔT68 (°C)	Upper shelf (J)	ΔRTNDT (°C)
Number of tests	145			574		96
	≤ 55 (1/4 T)	≤ 74 (1/4 T)	≥ 20	50<Δ<58 (1/4 T)	> 172	45 (1/4T)

- Toughness
  - Ductile tearing results are in conformance with RCC-M specifications
  - Fracture toughness : not all results are covered by RCC-M minimal value curve when indexed to acceptance tests RTNDT
  - A shift of 20 °C is necessary to cover the results



Results - SG lower shell

- Segregated area extent

Configuration	Max %C	Position
Discarded + machined	0.28	Between ½ T and int. ¼ T

  - Manufacturing data show that no segregated material is in contact with the fluid → no corrosion issue

- Mechanical properties

	Rp0,2 (Mpa)	Rm (Mpa)	A %	ΔT68 (°C)	ΔRTNDT (°C)	Toughness
Number of tests in segregated area	30			183	48 drop weight tests	150 CT(0.5T)
	In conformance with requirements			45	10	All results are covered by RCC-M indexed to acceptance tests RTNDT

- HAZ properties (1 mm and 4 mm to fusion line)
  - No deviation to mechanical properties requirements
  - Implant tests showed no risk of cold cracking on segregated material

---

Results

- ↗ Mechanical properties in fast fracture studies were based on these results

- ↗ The difference of DBTT shift between each justification file is linked to the localization of the segregated area, i.e distance to quenched surfaces

Global view

	Justifications issued by EDF	Justifications accepted by ASN	Sacrificial components characterization	Safety body decision
RV heads	☑	☑	☑	☑
SG lower shell	☑	☑	☑	☑
SG Channel heads	☑	☑	ongoing	
SG tube sheets	☑	ongoing		

Content

- Context and safety issues
- Manufacturing conditions
- Justification data
- IRSN position

IRSN position

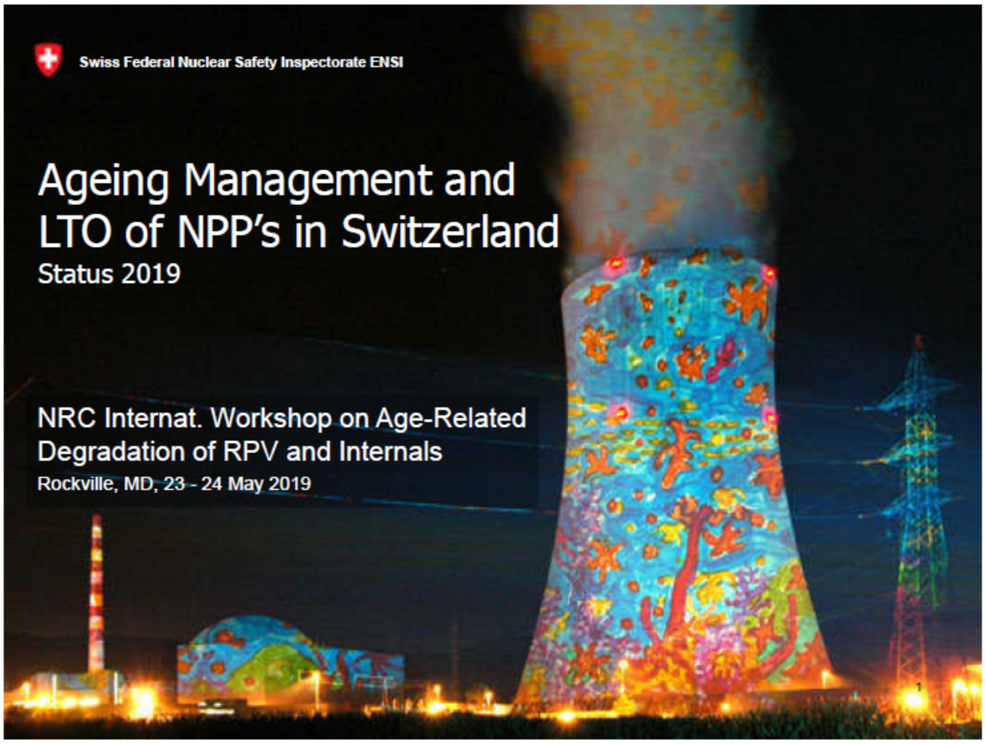
- Justification method based on sacrificial components
  - Necessary approach as calculation approach would raise uncertainties and representativity issues
  - Representativity is key
  - Additional conservatism on material properties to integrate representativity and variability issues

- Type of tests supporting the demonstration
  - IRSN recommends the use of drop weight tests in addition to CVN and CT tests
    - CT tests are necessary but not sufficient especially to justify the integrity of components whose manufacturing process has little OPEX (hollow ingot for instance)
  - CT tests provide information at a local scale on heterogeneous materials
  - Drop weight tests provide information at a mesoscopic scale

↗ Both types of tests are complementary

Thanks for your attention

B.23 Aging Management and LTO of NPPs in Switzerland: Status 2019 (R. Doering)



Swiss Federal Nuclear Safety Inspectorate ENSI

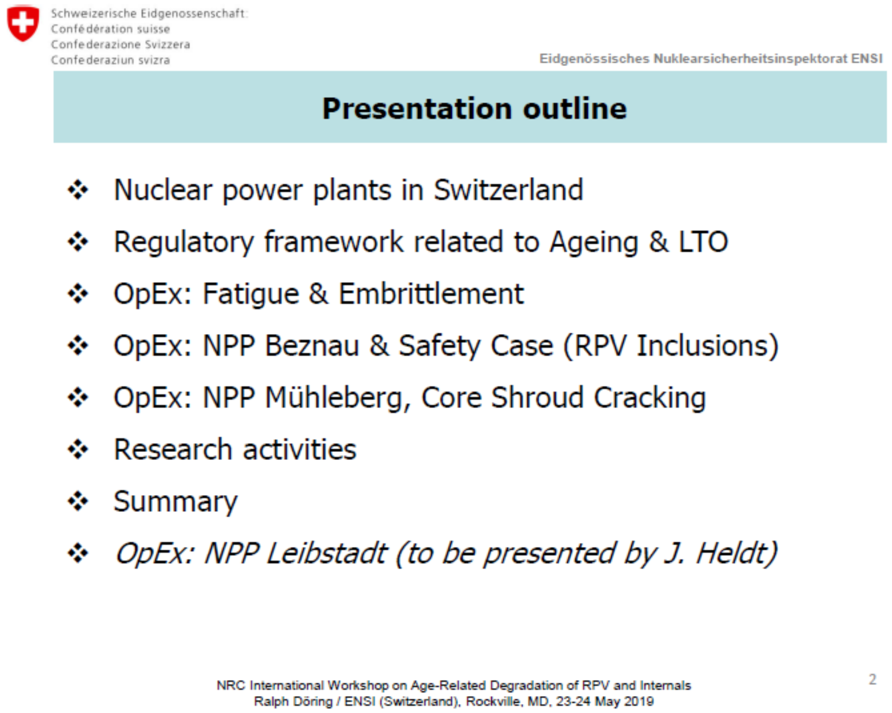
Ageing Management and LTO of NPP's in Switzerland

Status 2019

NRC Internat. Workshop on Age-Related Degradation of RPV and Internals

Rockville, MD, 23 - 24 May 2019

1



Schweizerische Eidgenossenschaft
Confédération suisse
Confederazione Svizzera
Confederaziun svizra

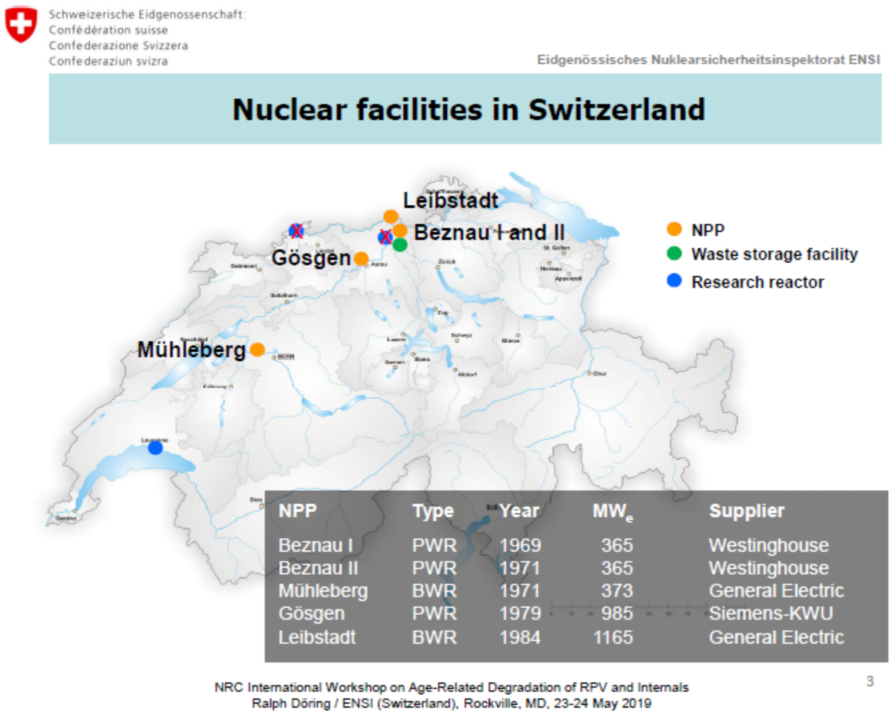
Eidgenössisches Nuklearsicherheitsinspektorat ENSI

Presentation outline

- Nuclear power plants in Switzerland
- Regulatory framework related to Ageing & LTO
- OpEx: Fatigue & Embrittlement
- OpEx: NPP Beznau & Safety Case (RPV Inclusions)
- OpEx: NPP Mühleberg, Core Shroud Cracking
- Research activities
- Summary
- *OpEx: NPP Leibstadt (to be presented by J. Heldt)*

NRC International Workshop on Age-Related Degradation of RPV and Internals
Ralph Döring / ENSI (Switzerland), Rockville, MD, 23-24 May 2019

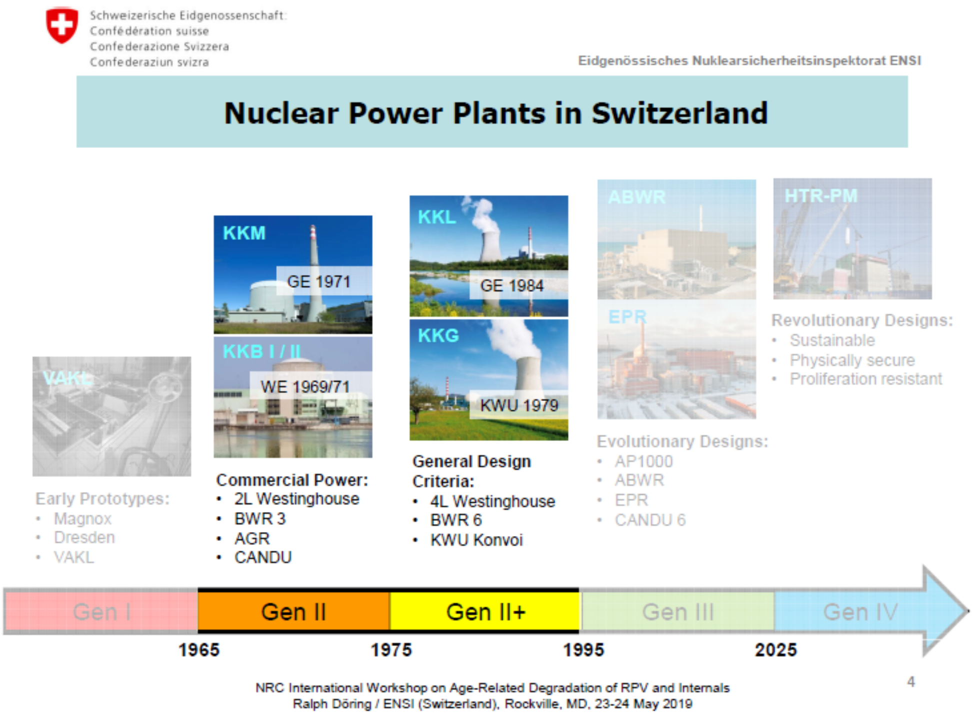
2



Nuclear facilities in Switzerland

● NPP
● Waste storage facility
● Research reactor

NPP	Type	Year	MW_e	Supplier
Beznau I	PWR	1969	365	Westinghouse
Beznau II	PWR	1971	365	Westinghouse
Mühleberg	BWR	1971	373	General Electric
Gösgen	PWR	1979	985	Siemens-KWU
Leibstadt	BWR	1984	1165	General Electric



Nuclear Power Plants in Switzerland

Early Prototypes:
- Magnox
- Dresden
- VAKL

Commercial Power:
- 2L Westinghouse
- BWR 3
- AGR
- CANDU

General Design Criteria:
- 4L Westinghouse
- BWR 6
- KWU Konvoi

Evolutionary Designs:
- AP1000
- ABWR
- EPR
- CANDU 6

Revolutionary Designs:
- Sustainable
- Physically secure
- Proliferation resistant

Nuclear Power Plants in Switzerland

	KKB 1	KKB 2	KKG	KKL	KKM
Thermal output [MW]	1130	1130	3002	3600	1097
Net electric output [MW] (2017)	365	365	1010	1220	373
Reactor supplier	WE	WE	KWU	GE	GE
Reactor type	PWR	PWR	PWR	BWR	BWR
Reactor model	1st Gen. 2 loop	1st Gen. 2 loop	2nd Gen. 3 loop (Pre-Konvoi)	BWR-6 GE-5	BWR-4 GE-3
Main heat sink	River water	River water	Cooling tower	Cooling tower	River water
Commenced commercial operation	1969	1972	1979	1984	1972
Final shutdown (assumption)	(2029 … 2031)	(2029 … 2031)	(≥ 2039)	(≥ 2044)	2019
Expected lifetime	(≈ 60)	(≈ 60)	(≥ 60)	(≥ 60)	48

Current LTO Status, 2019

Unlimited operating life time for NPP's but **conditions for preliminary taking out of service, Periodic Safety Review + LTO safety evaluation (10 year interval)**

1st generation NPPs in Switzerland

- Beznau-1 (1969), LTO approved for the period 2008 until 2018
  Beznau-2 (1971), LTO approved for the period 2010 until 2020
  second LTO (Unit 1&2) project submitted to ENSI in 2018 for the next 10 year period
- Mühleberg (1972), LTO approved until shut down in 2019

2nd generation NPPs in Switzerland

- Gösgen (1979), LTO project submitted to ENSI end of 2018
- Leibstadt (1984), LTO project in 2023 / 2024

LTO Requirements

After 40 years of operation:

- Unlimited operating life time for NPP's but **conditions for preliminary taking out of service**
- **Continued backfitting measures** to improve nuclear safety or reduce nuclear risks
- Periodic Safety Review + **LTO safety evaluation based on licensees' safety cases**

  2010, Report on Life Management for LTO NPP Beznau
  2012, Report on Life Management for LTO NPP Mühleberg

Time Limited Ageing Analysis (TLAA) & Safety Analyses

- RPV Integrity (irradiation surveillance program, neutron fluence assessment, PTS analyses)

  (ENSI-B01, Reg.-Guide 1.99 Rev. 2, KTA 3203)
- Pressure vessels and piping, mechanical and thermo-mechanical fatigue analysis

  (ENSI-B01,10 CFR 50.54, NUREG-1800, NUREG-6909)
- Pressure vessels and piping, leak before break analysis

  (NUREG 0800 SRP 3.6.3, NUREG-1061, Reg.-Guide 1.45
  KTA3206)
- Safety Analyses for NSSS
- Steel containment, corrosion monitoring and corrosion assessment
- Structural analysis of the concrete parts of the containment

Regulatory Timeline of AMP in Switzerland

- 1991 HSK enforced requirement for AMP
- 2001 RegGuide HSK-R-48 – Periodic Safety Review (PSR)
- 2004 RegGuide HSK-R-51 – Ageing Management (AMP)
- 2005 Nuclear Energy Act, Nuclear Energy Ordinance
- 2008 DETEC Ordinance "Preliminary shut down of NPP"
- 2008 RegGuide ENSI-B02 – Periodic Reporting
- 2009 IAEA NS-G-2.12, Safety Guide for AMP
- 2011 RegGuide ENSI-B01 – Ageing Management
- 2012 IAEA OSART Mission (with LTO module) Mühleberg NPP, 2014 OSART follow-up visit
- 2014 RegGuide ENSI-A03 – PSR for NPP (incl. LTO),
- 2017 Swiss Participation on ENSREG Topical Peer Review 2017,
- 2018 IAEA SSG-48, New Safety Guide for AMP -> Review of RegG B01
- 2018 ENSI guide manual for AM of storage tanks

Key legal documents and guidelines, NEA, NEO

- Nuclear Energy Act
  - ✓ Art. 22: Maintenance, PSR, AMP and Backfitting
- Nuclear Energy Ordinance
  - ✓ Art. 32 Maintenance, in-service inspection and functional testing,
  - ✓ Art. 33 Systematic safety and security assessments
  - ✓ Art. 34 Comprehensive safety review for nuclear power plants
  - ✓ Art. 35 Ageing management
  - ✓ Art. 36 Monitoring the state of the art in science and technology and the operating experience in comparable installations
  - ✓ Art. 44 Conditions for the taking out of service and backfitting of nuclear power plants

Key legal documents and guidelines, Ordinances

- ➢ Ordinance for safety classified vessels and piping
  - ✓ During design, fabrication and construction of mechanical components materials with highest quality shall be used to minimize the impact of ageing effects
- ➢ DETEC Ordinance “Preliminary shut down of NPP”

A NPP has to shut down if the integrity of the physical barriers is compromised by ageing mechanisms (definition of acceptance criteria)

- ✓ RPV embrittlement: - adjusted reference temperature > 93 °C
  - upper shell Charpy energy < 68 J
- ✓ Wall thickness of primary piping < original design thickness
- ✓ Cracking of primary piping < allowable limits (fracture mechanics)
- ✓ Wall thickness of primary containment < original design thickness

For LTO: Demonstrate compliance with these criteria for ageing over the expected operation time by TLAA

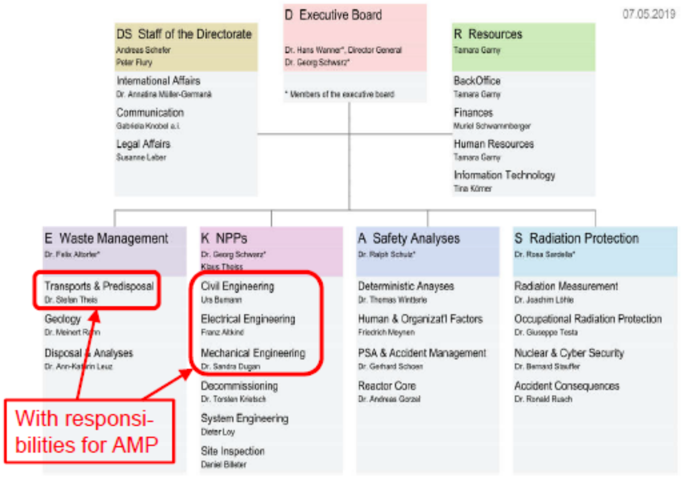
Key legal documents and guidance, RegG ENSI-B01

- ➢ Guideline ENSI-B01 (specific requirements for AMP)
  - ➢ Scope of the SSCs to be covered in the AMP
    - ✓ All safety classified structures and electrical components
    - ✓ All safety class 1-3 vessels and piping and related mechanical components (e.g. pumps, valves, supports and attachments)
  - ➢ Methods used for the verification of the embrittlement resistance of the RPV (Mandatory TLAA, Appendix 5)
  - ➢ Scope of the components to be covered in the fatigue monitoring and assessment program (Mandatory TLAA, Appendix 6)
- ➢ AMP basic documents (to prepare by licensees)
  - ➢ Plant-specific or generic guidelines (GSKL, Swiss NPP owners group),
  - ➢ KATAM (since 1991, periodical update): compendium of potential materials ageing mechanisms for mechanical Equipment in NPP (GSKL),
  - ➢ Fact sheets (“Steckbriefe”): basic data, tables of ageing mechanisms relating to the affected components, component history, periodical update (licensee),

Key legal documents and guidance, RegG ENSI-B02, -A03

- Guideline ENSI-B02 (annual reporting)
    - Overview of changes performed in the AM-documents
    - Results of the fatigue surveillance program
    - Results of the review of the operating experience and the follow-up of the current state of science and technology
    - Description of the consequences of these results on the AMP
    - Evaluation of the efficiency of the AMP based on failure statistics and maintenance indications
- Guideline ENSI-A03 (additional reporting within the PSR)
    - Review of the qualification status of the in-scope SSCs
    - Assessment of technological obsolescence
    - Review of existing time limited ageing analyses with respect to LTO

The regulatory body ENSI (2019)



Swiss Federal Safety Inspectorate ENSI at all has around 150 employees

Mechanical Engineering Group 8 (6) employees

Supported by SVTI (TSO)

with 26 employees

SVTI reviews and supervising In-service inspections, manufacturing, installation and repair activities.

NDT Qualification body

Main degradation mechanisms

- Fatigue (LCF, HCF, TF, TMF, EAF)
- Irradiation Embrittlement
- Stress Corrosion Cracking (SCC)
- FAC
- Other (corrosion, wear, erosion, thermal ageing, …

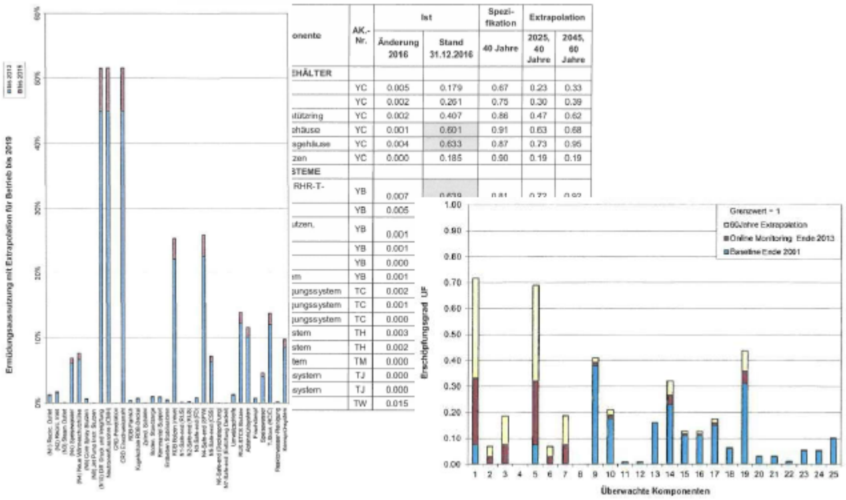
Fatigue Monitoring in Swiss NPP (1)

- 3 of 4 Swiss NPP using fatigue monitoring systems
  - KKL: “Fatigue Pro”, since 2007 (installed 2003), EPRI / Structural Integrity
  - KKG: “FAMOSi”, since 2014, Framatome
  - KKB: “WESTEMS”, since 2002, Westinghouse
  - KKM: Manual recording and assessment of fatigue
- Selection of monitoring positions based on recommendations and experience of manufacturer
- Typically around 25 - 30 positions are permanently monitored, including nozzles of RPV, steam generators and pressurizer and some piping like main cooling line and surge line.

Fatigue Monitoring in Swiss NPP (2)

- Measurement of temperature-time histories to identify transients (stress based approach)
- Baseline fatigue (before installing monitoring systems) is estimated from counted transients based on design specification (cycle based approach)
- Environmental effects on fatigue have to be considered according to ENSI-B01 and NUREG/CR-6909
- Annual fatigue report submitted to regulator about the current and extrapolated fatigue usage factors for 60 operation years (or EOL).

Fatigue Monitoring, annual reporting



onente	AK.-Nr.	Ist		Spezi-fikation	Extrapolation	
		Änderung 2016	Stand 31.12.2016	40 Jahre	2025, 40 Jahre	2045, 60 Jahre
EHÄLTER						
	YC	0.005	0.179	0.67	0.23	0.33
	YC	0.002	0.261	0.75	0.30	0.39
stützring	YC	0.002	0.407	0.86	0.47	0.62
ehäuse	YC	0.001	0.601	0.91	0.63	0.68
agehäuse	YC	0.004	0.633	0.87	0.73	0.95
zen	YC	0.000	0.185	0.90	0.19	0.19
STEME						
RHR-T-	YB	0.007	0.639	0.81	0.72	0.92
	YB	0.005				
utzen.	YB	0.001				
	YB	0.001				
	YB	0.000				
em	YB	0.001				
jungssystem	TC	0.002				
jungssystem	TC	0.001				
jungssystem	TC	0.000				
stem	TH	0.003				
stem	TH	0.002				
tem	TM	0.000				
system	TJ	0.000				
system	TJ	0.000				
	TW	0.015				

Fatigue Monitoring, backfittings & modifications

- KKM
  - 1995 to 1996, the RPV head bolts were replaced, replacement bolts having rolled threads and optimized thread geometry (calculated fatigue life 90 Y, bolts are the leading position for RPV)
  - In 1997, based on results from the fatigue monitoring measures, the RPV nozzles were modified by the installation of double thermo-sleeves.
  - A blind flange was fixed on RPV nozzle N9 in 2004. This modification largely eliminated fatigue loadings on this nozzle.
- KKG
  - 2005 PISA-project (TMI follow-up action), backfitting of primary pressure relief system, reduction of fatigue usage factors on surge line and pressurizer nozzles due to these system modifications and replacements of some pipe-sections
- KKB
  - 2013 New shafts for primary reactor water pumps (new material, optimized geometry, modified seal water flow)

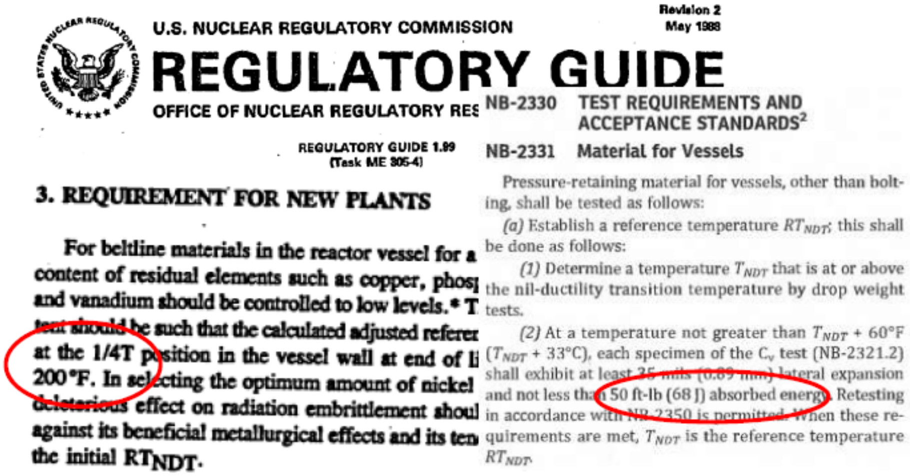
Fatigue Monitoring, conclusions

- The results of the fatigue monitoring in the form of current fatigue usage factors and the corresponding levels extrapolated to 60 years of operation are submitted to ENSI in an annual report.
- The existing results show that the long-term operation of the Swiss NPP is not subject to any limitations as a result of RPV material fatigue.
- Potential future challenges:
  - Prevention of fatigue damages caused by vibrations (small piping, rotating equipment),
  - Load flexible operation

Program for assessment of irradiation embrittlement

- Neutron fluence assessment (Reducing of fluence by low leakage core configuration)
- Testing of surveillance specimens
- Determination of ART based on requirements of Guideline ENSI-B01 (U.S.-NRC 1.99 Rev. 2 / ASTM E 1921 Master Curve Concept)
- Embrittlement have to comply with “DETEC Criteria” (ART < 93°C in ¼ wall depth, Charpy energy ≥ 68 J)
- PTS analyses by deterministic FM (PWR)
- P-T limiting curves

Background of ”DETEC Criteria” for RPV embrittlement



Revision 2
May 1988

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RES

REGULATORY GUIDE 1.99
(Task ME 305-4)

3. REQUIREMENT FOR NEW PLANTS

For beltline materials in the reactor vessel for a
content of residual elements such as copper, phos
and vanadium should be controlled to low levels.* T
test should be such that the calculated adjusted referer
at the 1/4T position in the vessel wall at end of li
200°F. In selecting the optimum amount of nickel
deleterious effect on radiation embrittlement shoul
against its beneficial metallurgical effects and its ten
the initial RT_{NDT}.

NB-2330 TEST REQUIREMENTS AND ACCEPTANCE STANDARDS[2]

NB-2331 Material for Vessels

Pressure-retaining material for vessels, other than bolting, shall be tested as follows:

(a) Establish a reference temperature RT_{NDT}; this shall be done as follows:

(1) Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

(2) At a temperature not greater than T_{NDT} + 60°F (T_{NDT} + 33°C), each specimen of the C_v test (NB-2321.2) shall exhibit at least 35 mils (0.89 mm) lateral expansion and not less than 50 ft-lb (68 J) absorbed energy. Retesting in accordance with NB-2350 is permitted. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

Status of surveillance programmes, March 2019

4 reactors have finished the surveillance programmes.
Continued in Leibstadt NPP, 3th capsule will be removed in 2020.

	KKB1	KKB2	KKG	KKL	KKM
Tested specim.	6	5	3	2	3*
last removed	2010	2010	2006	2008	2002
Coverage by tested specim.	> 60 Y	> 60 Y	> 100 EFPY	> 24 Y	> 30 Y (BM, v3) 78 EFPY (weld v2)
Remaining spec. in RPV	-	1	-	5	1* (reinstalled in 2004)
coverage 2019	-	> 80 Y	-	> 35 Y	≈ 50 Y (BM, v3)

* All specimens provided to PSI for future research activities.

Adjusted reference temperatures for 54 EFPY / 60 Y

	KKB1	KKB2	KKG	KKL	KKM
ART [°C] Method I inner wall / 1/4T	104 / 98	76 / 70	33 / 28	9	61 (V2) 17 (BM)
RT_{ref} [°C] Meth. IIA inner wall / 1/4T	80 / 74	51 / 46	-16 / -18		
Min. upper shelf Charpy energy [J]	137	120	118	> 119	125
Material	BM Ring C	BM Ring C	BM II	BM	Weld V2 BM

(KKG has installed ex-vessel dosimetry.
It allows a more precise fluence calculation.)

Swiss Regulatory Guide ENSI-B01, Appendix 5

Alternative assessment methods for RT_{ref} :

- Method I (US NRC Regulatory Guide 1.99 Rev. 2):
  RT_{ref} = ART
- Method II-A:

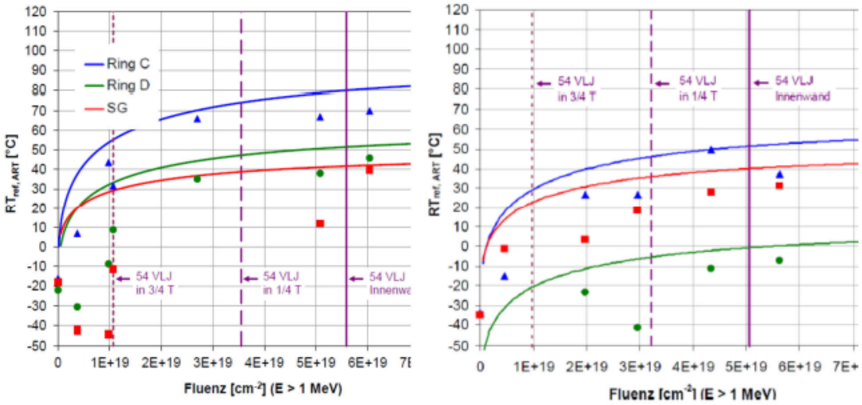
$$RT_{ref} = T_0 + 14.4K + \Delta T_S + \sqrt{324/n + 16 + \Delta T_M^2 + \Delta T_T^2}$$

T_o	reference temperature acc. to ASTM E 1921 (Master curve approach)
n	number of valid tests
ΔT_s	=0 for 1T-C(T) specimens; =10K for 0.4T-SEN(B) specimens
ΔT_M	=0 for base material; =6K for weld material
ΔT_T	=0 for 1T-C(T) specimens; =5K for 0.4T-SEN(B) specimens

- Method II-B:
  $RT_{ref} = RT_{ref}^{(o)} + \Delta T_j$

$RT_{ref}^{(o)}$	reference temp. acc. to equation of method II-A for unirradiated material
ΔT_j	temperature shift according to NRC Regulatory Guide 1.99 Rev. 2

Embrittlement KKB 1&2, ENSI-B01 Method-IIA (MC)



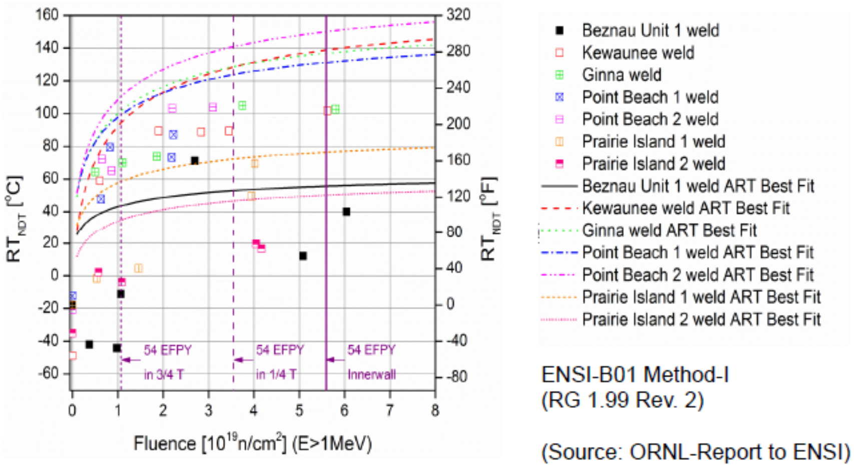
KKB, Unit 1

KKB, Unit 2

Embrittlement KKB 1, Ring C, ENSI-B01, 54 EFPY / 60 Y

Material	RPV position	Neutron fluence [cm^{-2}]	RT$_{ref}$ [°C]	
Method I Charpy	surface	5.59E+19	104	
	¼ wall thickness	3.55E+19	98	
Method II-A Master Curve	surface	5.59E+19	80	
	¼ wall thickness	3.55E+19	74	DETEC ≤ 93°C
Method II-B T_0+Charpy-shift	surface	5.59E+19	89	
	¼ wall thickness	3.55E+19	83	DETEC ≤ 93°C

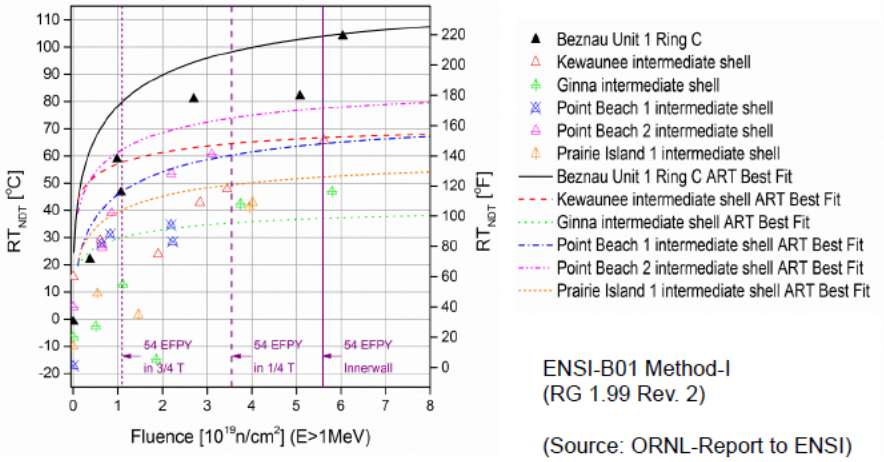
RT$_{NDT}$ and the ART best fit curves for Beznau Unit 1 core weld, compared to Westinghouse US PWRs of same design



ENSI-B01 Method-I
(RG 1.99 Rev. 2)

(Source: ORNL-Report to ENSI)

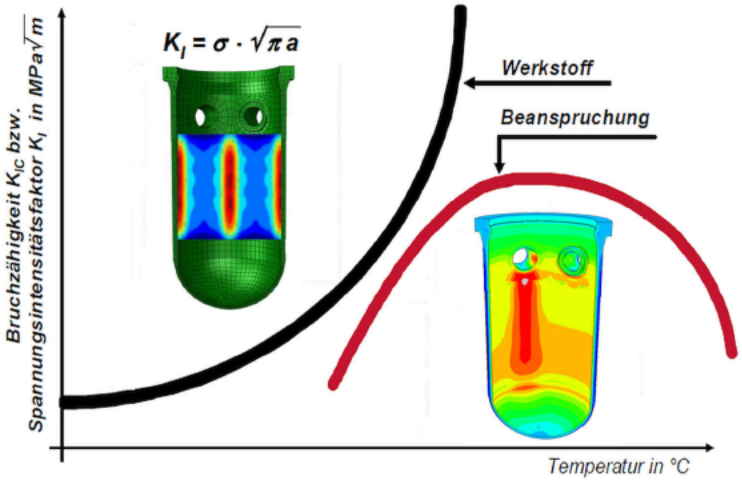
RT_{NDT} and the ART best fit curves for Beznau Unit 1 Ring C, compared to Westinghouse US PWRs of same design



ENSI-B01 Method-I
(RG 1.99 Rev. 2)

(Source: ORNL-Report to ENSI)

PTS-Analyses based on deterministic FM



PTS-Analyses based on deterministic FM

For PWR's a PTS-Analysis is required which include following tasks:

- Thermo-hydraulic Analyses (RELAP5, KWU-Mix, optional CFD) for a comprehensive load case matrix:
  - different leak sizes from 3 cm^2 up to 2F-break,
  - leak locations in hot and cold leg, different injection temperatures,

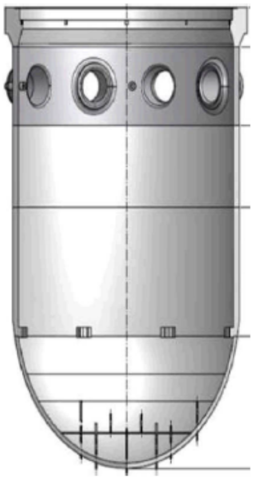
  (For Swiss NPP the medium size LOCA's are leading.)
- Fracture Mechanics Analyses for:
  - Different crack locations (Nozzle corner, **core region**),
  - Different crack orientations (axial / **circumferential**),
  - Different crack types (surface / undercladding),
  - Different crack depths, aspect ratio is given by code (1/5 or 1/6)
  - Possible options with/without WPS,
  - Additional options, but not used for Swiss NPP PTS analyses: constraint effects, crack arrest, PFM to demonstrate conservatisms

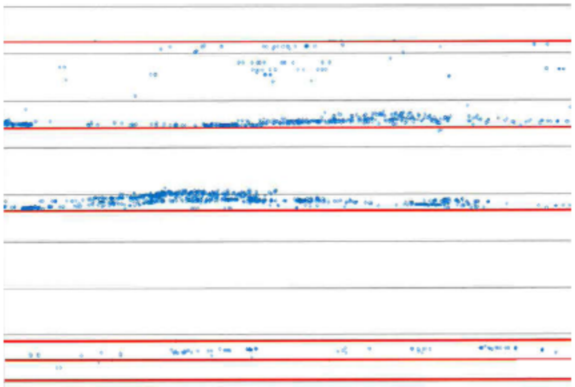
PTS-Analyses for Beznau and Gösgen NPP based on KTA 3203 methology.

Irradiation embrittlement, conclusion

- The results of the surveillance programs in the form of fluence calculations and the corresponding levels of irradiation embrittlement extrapolated to 60 years of operation are submitted to ENSI.
- Highest embrittlement of all Swiss Reactors have the Base Material of Ring C in Beznau unit 1. It fulfils the requirements of 93°C shut down criteria of the DETEC Ordinance.
- The existing results show that the long-term operation of the Swiss NPP is not subject to any limitations as a result of RPV irradiation embrittlement.
- PTS-Analyses showed for all Swiss PWR that the allowable $RT_{,PTS}$ (with consideration of WPS effect) is higher than the 93°C DETEC shut down criteria.

KKB 1, UT Indications in 2015

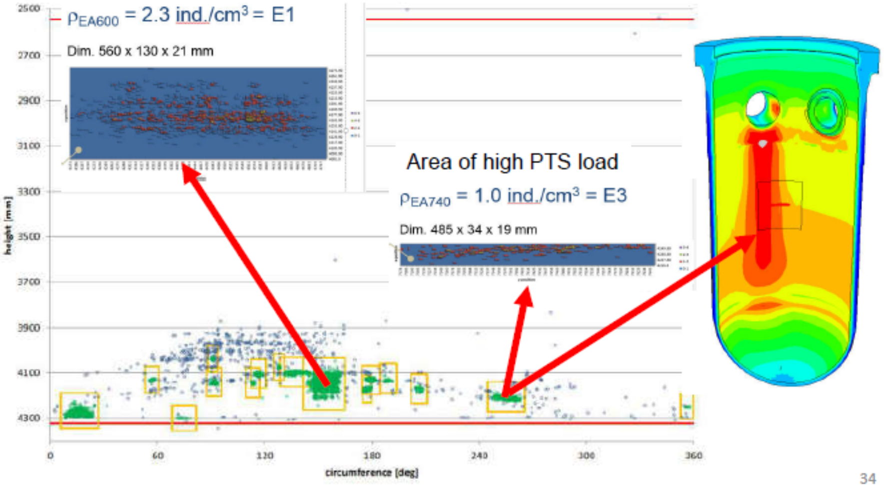




- Indications identified as Al2O3-Inclusions
- Not Ageing, manufacturing problem (insufficient discard rate from ingot)

KKB 1, Safety Case

Area of high density of inclusions



ρ_{EA600} = 2.3 ind./cm³ = E1

Dim. 560 x 130 x 21 mm

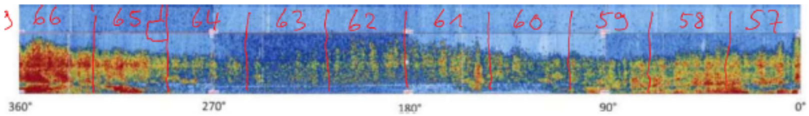
Area of high PTS load

ρ_{EA740} = 1.0 ind./cm³ = E3

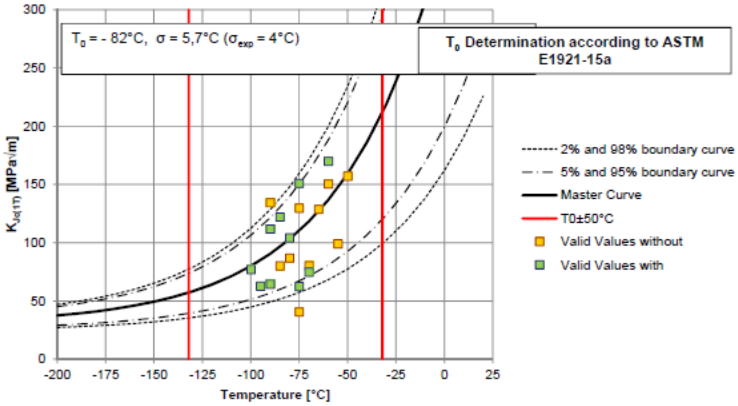
Dim. 485 x 34 x 19 mm

KKB 1, Replica of Ring C based on manufacturing records

metallurgical bottom



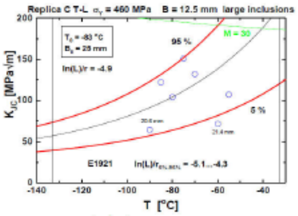
KKB 1, Master Curve by 1 CT-Specimen from Replica



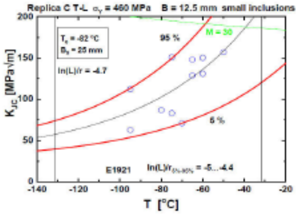
Master Curve Testing of the replica shell C material: Specimens with inclusions show no difference to specimens without inclusions

KKB 1, Master Curve by 0.5 CT-Specimen from Replica

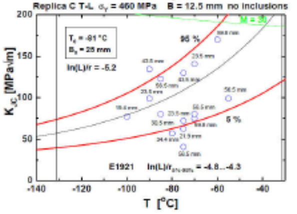
large inclusions of 2mm on process zone



inclusions < 2mm on process zone



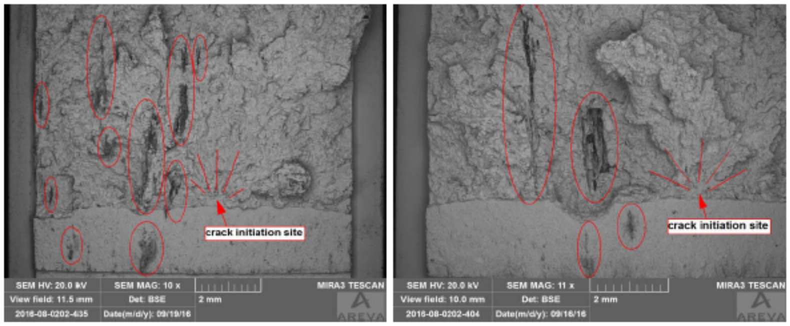
no inclusions on process zone



A standard MC analysis indicate that specimens with large or small inclusions in the fracture process zone are homogeneous based on the log-likelihood criterion.

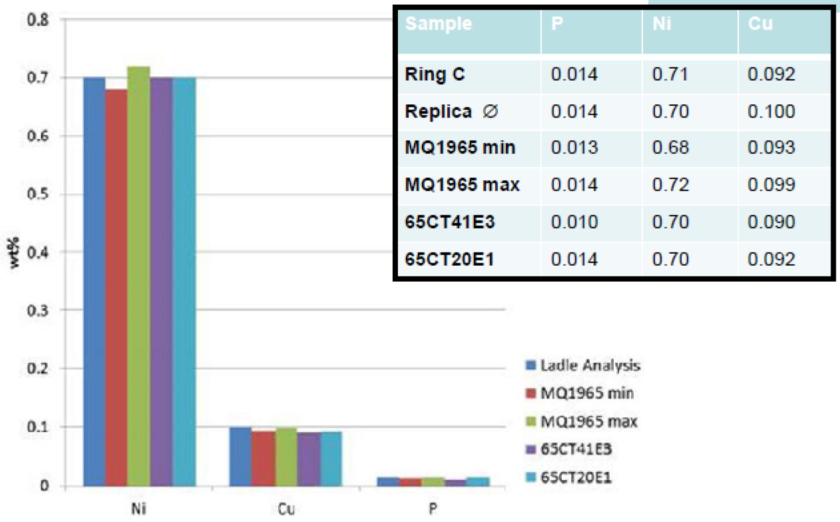
Only the specimens without inclusions show an indication of inhomogeneity.

KKB 1, Fractography of Replica Material



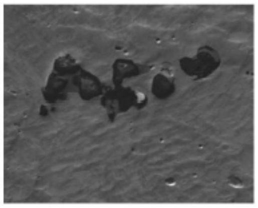
Fracture surface (SEM) of broken CT-specimens, replica C material
Result: No influence of Al_2O_3 inclusions on crack initiation

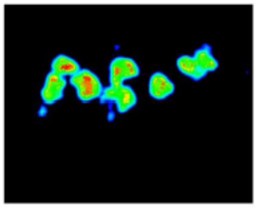
KKB 1, Comparison of Chemical Composition

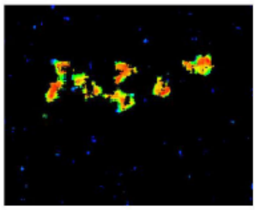


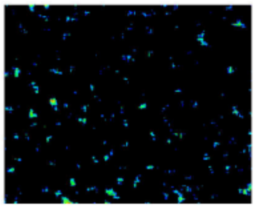
Sample	P	Ni	Cu
Ring C	0.014	0.71	0.092
Replica ∅	0.014	0.70	0.100
MQ1965 min	0.013	0.68	0.093
MQ1965 max	0.014	0.72	0.099
65CT41E3	0.010	0.70	0.090
65CT20E1	0.014	0.70	0.092

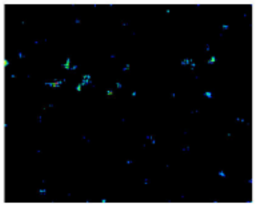
KKB 1, Local Chemical Composition near Inclusions

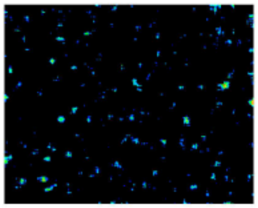

C7-B154 Inclusion 1, BSE


Al Kα, element distribution


O Kα, element distribution


Cu Kα, element distribution


P Kα, element distribution

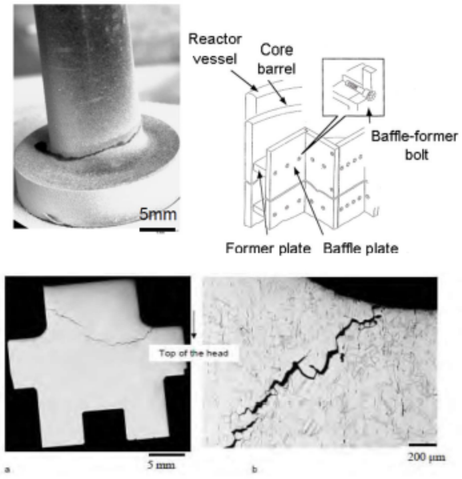

Ni Kα, element distribution

KKB 1, Safety Case Summary and Conclusions

- UT indications found in 2015 in RPV Base material were identified as non-metallic aluminium oxide inclusions.
- Fracture mechanics testing on specimen from replica ring C did not show an influence of these inclusions.
- Fractographic analyses never showed crack initiation at Al_2O_3-Inclusions.
- Inclusions do not have a negative effect on fracture toughness.
- According to chemical analyses, Inclusions do not influence irradiation embrittlement behaviour.
- Large inclusions or inclusion clusters, so called “high amplitude indications”, were conservatively assumed as flaws. These assumed flaws are covered by the ASME XI, IWB-3000 acceptance standards as well by the postulated cracks for PTS analyses.
- An alternative multiple flaw combination rule was proposed, now Code Case N-877. Acceptance criteria met with CC N-877 as well with standard rules according to Section XI, IWA-3000.

Examples of IASCC Incidents in LWRs

Baffle Former Bolts in PWR



Pokor et al., Fontevraud 6, A154-T02, 2006

- Replacement of baffle former bolts at Beznau unit 1&2 in 2009 and 2010
- 192 bolts (unit 2)
- 195 bolts (unit 1)
- Replacement in an optimized pattern
- new material (AISI 316 CW)
- Optimized notch contour to reduce stresses

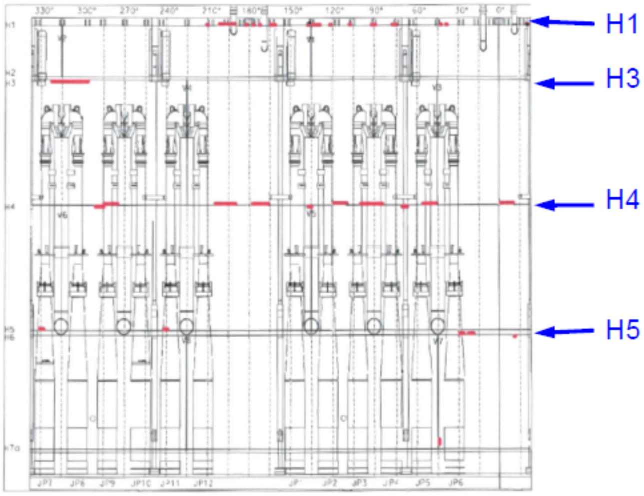
KKB, (HERA) 2015/16 RPV Head Replacement

- New RPV closure heads with a series of small improvements
- Inconel 600 replaced by Inconel 690 and 52 (head penetration seal welds),
- No control rod thermal sleeve wear issue (no thermal sleeve),
- A small boric acid cavity at unit 1 removed.

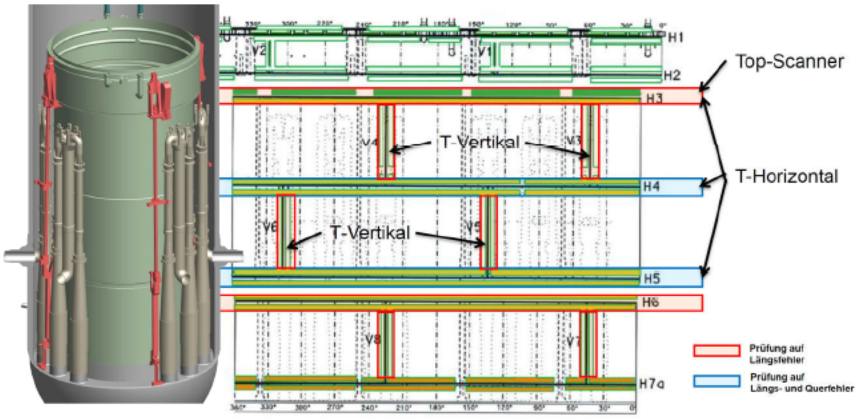
KKM, Overview core shroud cracking

- 1990 cracks in HAZ of H4 weld discovered by VT inspections, 1991 UT on H4, IGSCC driven by weld residual stresses, starting from inner surface
- 1992/93 conductivity of reactor water optimized (to around 0.1 µS/cm), 2" Plug sample removed
- 1993 on HSK request, start of qualified systematic (1 or 2-year) VT and UT (GE from ID)
- 1994 Simplified FM-model (screening criteria)
- 1996 installation of pull rods and radial stabilizers at 4 positions, precautionary safety action against unexpected fast crack growth. VT of rods
- 2000 HWC & NobleChem, 2005 OLNC (platinum injection),
- 2011 3D FE-model for FM analyses to LTO
- 2011 New UT-System (WE, from OD with measurement of depth), enhanced coverage
- 2013 New FE-model with circumferential cracks, 60 % wall thickness, acoustic loads, further optimization of Pt feed-in strategy
- 2014 replacement of pull rods planned, 6 rods new design, cancelled because of shut down
- 2014 first detection of transversal cracks (off-axis flaws), FE-model adapted
- 2015 & 2016 inspections H4 (UT, VT)
- 2018 last inspection (VT, H4 and the off axis cracks), no propagation

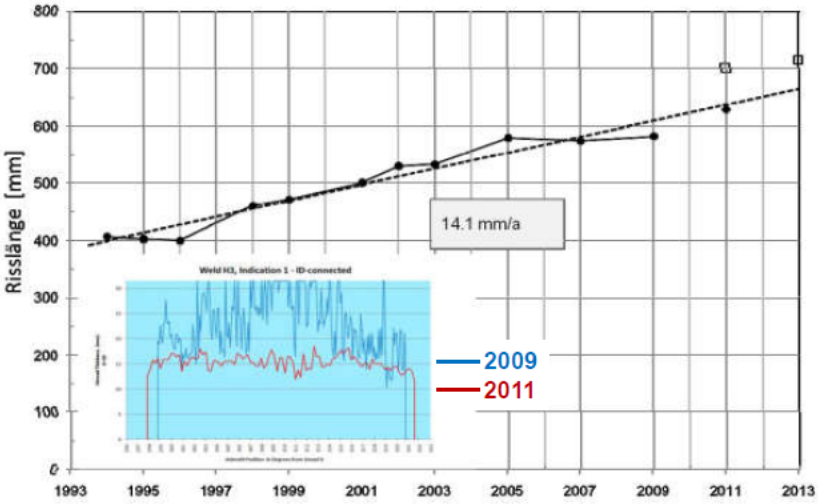
KKM, Core Shroud Cracks, 2011



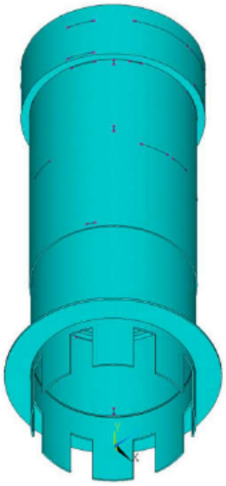
KKM, systematic Core Shroud measurements since 1993

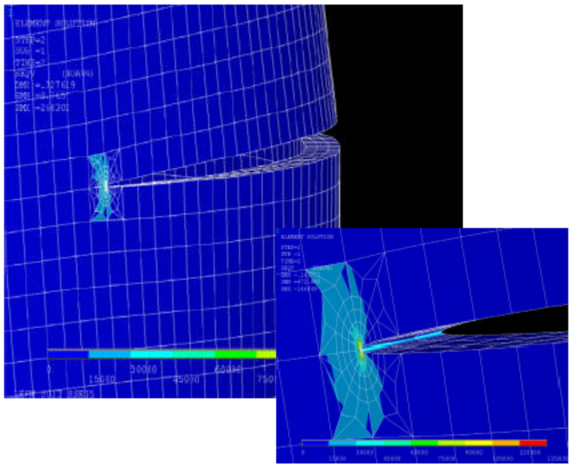


KKM, Core Shroud Crack, Indication 1, Weld H3



KKM, Core Shroud Model with through wall cracks, 2011





KKM, Results of FM analyses of core shroud cracks

LC 6.3.1: SSE von 102°, 3 Zuganker, Δp Recirc-Bruch, Eigengewicht, ΔT

	$K_{I\,max}$ [MPa√m]	K_{Ic} [MPa√m]	SF_{erf}	SF	Marge
H1	15.8	165	1.39	10.4	> 3 x
H2	33.9	165	1.39	4.9	> 3 x
H3	41.4	165	1.39	4.0	2.9 x
H4	39.5	123	1.39	3.1	2.2 x
H5	---	165	1.39	---	---
H6	24.7	165	1.39	6.7	> 3 x
V3	15.1	123	1.50	8.1	> 3 x
V4	3.9	123	1.50	> 15	> 3 x
V7	25.7	165	1.50	6.4	> 3 x

SSE according to "PEGASOS" (increased seismic hazards), 2013 acoustic loads added

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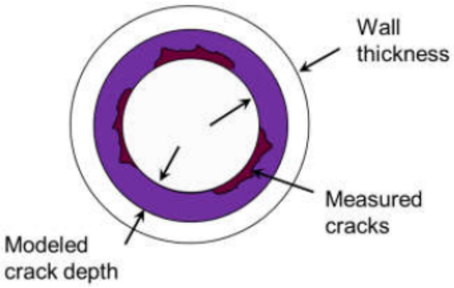
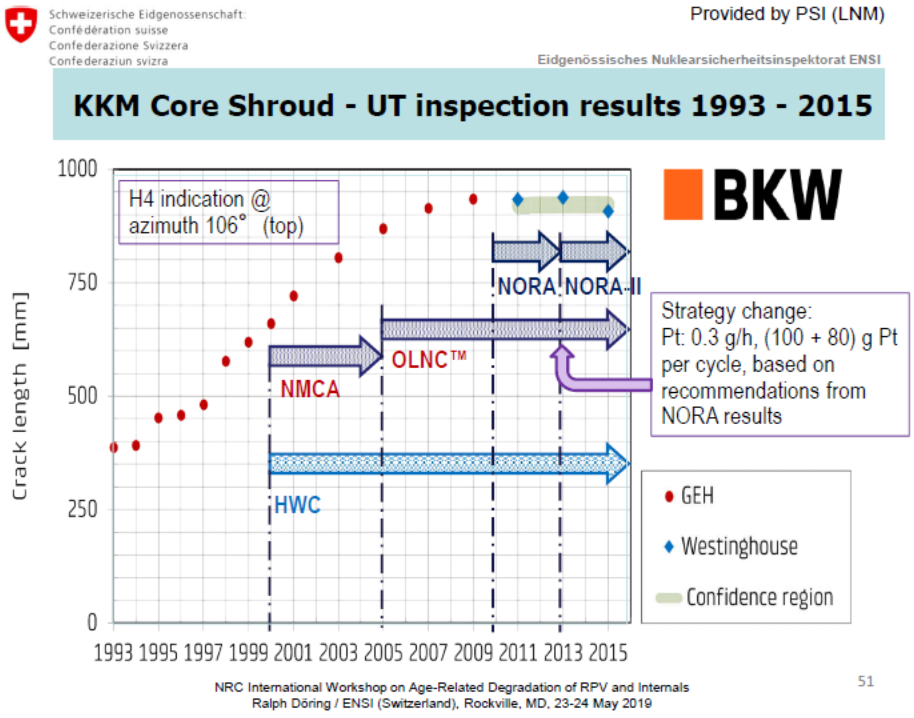
KKM, Core Shroud Model with circumferential cracks, 2013



Modelling:
3D-Finite Element Modeling of whole Core Shroud:
Enveloping 360° cracks at H3 and H4 with crack depth of 75% resp. 66% of wall thickness.

Acceptance Criteria based on LEFM: (defined by ENSI (Swiss regulator) for the specific KKM situation considering PCO 2019)
- KI max < 75 MPa√m (transversal cracks);
- ℓ transversal through-wall cracks < 320 mm



Schweizerische Eidgenossenschaft
Confédération suisse
Confederazione Svizzera
Confederaziun svizra

Provided by PSI (LNM)

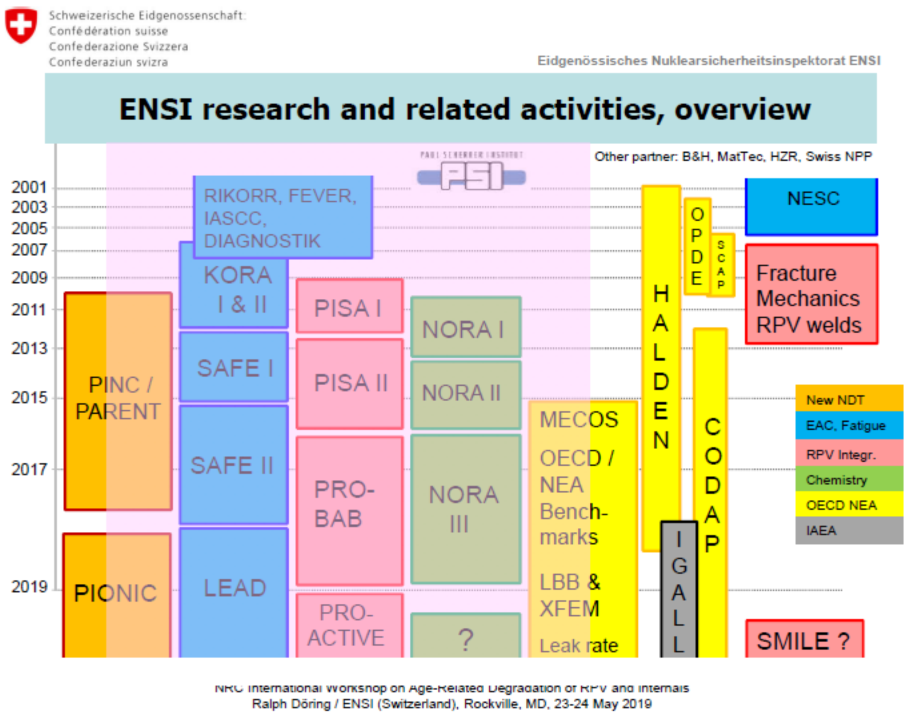
Eidgenössisches Nuklearsicherheitsinspektorat ENSI

KKM Core Shroud - UT inspection results 1993 - 2015

Strategy change: Pt: 0.3 g/h, (100 + 80) g Pt per cycle, based on recommendations from NORA results

NRC International Workshop on Age-Related Degradation of RPV and Internals
Ralph Döring / ENSI (Switzerland), Rockville, MD, 23-24 May 2019

51



Schweizerische Eidgenossenschaft
Confédération suisse
Confederazione Svizzera
Confederaziun svizra

Eidgenössisches Nuklearsicherheitsinspektorat ENSI

ENSI research and related activities, overview

Other partner: B&H, MatTec, HZR, Swiss NPP

RIKORR, FEVER, IASCC, DIAGNOSTIK

KORA I & II

SAFE I

SAFE II

LEAD

PINC / PARENT

PIONIC

PISA I

PISA II

PRO-BAB

PRO-ACTIVE

NORA I

NORA II

NORA III

?

MECOS

OECD / NEA Bench-marks

LBB & XFEM

Leak rate

HALDEN

IGALL

OPDE

SCAP

CODAP

NESC

Fracture Mechanics RPV welds

SMILE ?

- New NDT
- EAC, Fatigue
- RPV Integr.
- Chemistry
- OECD NEA
- IAEA

NRC International Workshop on Age-Related Degradation of RPV and Internals
Ralph Döring / ENSI (Switzerland), Rockville, MD, 23-24 May 2019

PROBAB (2016-18, ENSI): RPV Structural Integrity

- **SP-I (25 %): Systematic LOCA transients study by LTH and LRS**
  - Analysis of various LOCA scenarios (T, p, dm/dt, h) with system (LRS) and CFD codes (LTH)
  - Evaluation of cooling for various TH regimes (single & two-phase, plume & stripe cooling)
  - Uncertainties of transients & critical evaluation of transient matrix
- **SP-II (50 %): Probabilistic & deterministic PFM PTS analysis of a reference RPV by LNM**
  - Various LOCA and other transients (LTOP, …)
  - Consideration of plume & stripe cooling by engineering models
  - Material inhomogeneity's (segregation, inclusion bands) and hydrogen flakes
  - Effect of cladding & residual stress (cladding, welds)
  - 3D FEM & XFEM analysis for various crack configurations, validation of XFEM
- **SP-III (25 %): Probabilistic integrity, lifetime & LBB assessment for material ageing by LNM**
  - Literature survey for active ageing mechanism (SCC, EAF, TMF) → state-of-the-art
  - Evaluation of potential codes (PRO-LOCA, X-LPRM, …) and selection of code → POR-LOCA
  - Application of code to a selected specific case (e.g., KKL feedwater nozzle) in a first pre-study
  - Participation in OECD XFEM & LBB benchmarks / round robins study & in PARTRIDGE project

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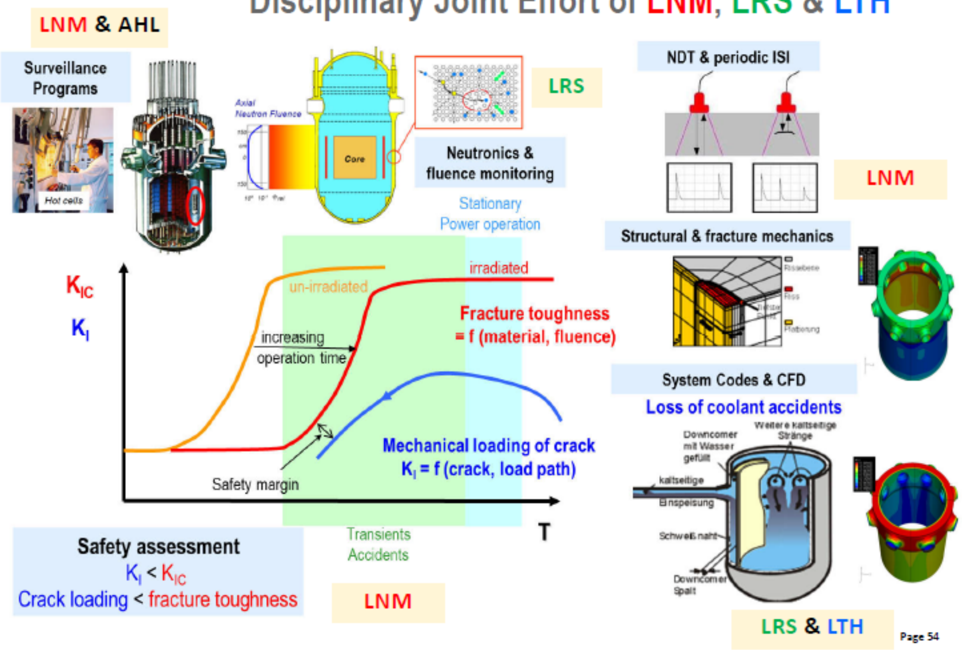
PROBAB: RPV Integrity Assessment as Multi-Disciplinary Joint Effort of LNM, LRS & LTH



LNM & AHL

Surveillance Programs

LRS

Neutronics & fluence monitoring

Stationary Power operation

NDT & periodic ISI

LNM

Structural & fracture mechanics

K_{IC}

K_I

Fracture toughness = f (material, fluence)

Mechanical loading of crack K_I = f (crack, load path)

Transients Accidents

System Codes & CFD

Loss of coolant accidents

Safety assessment
$K_I < K_{IC}$
Crack loading < fracture toughness

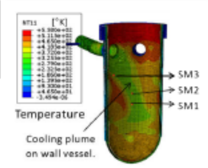
LNM

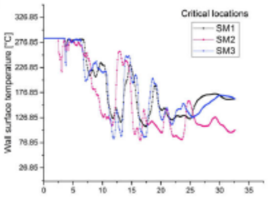
LRS & LTH

3D SM & FM Simulations of LOCAs by LNM

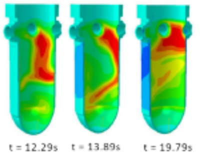
LBLOCA

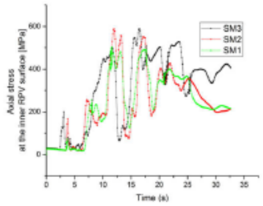
Wall temperature distribution

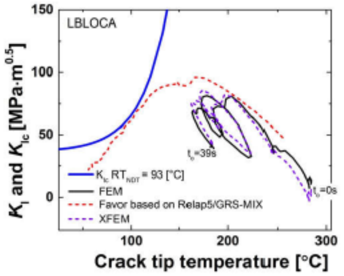


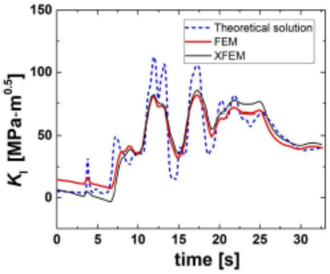


Axial stress distribution









Peak K_I at crack-tip in plume may be higher than in simplified 1D-models

Complex time course with significant reloading → WPS may be challenged

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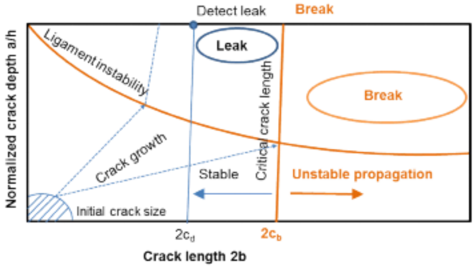
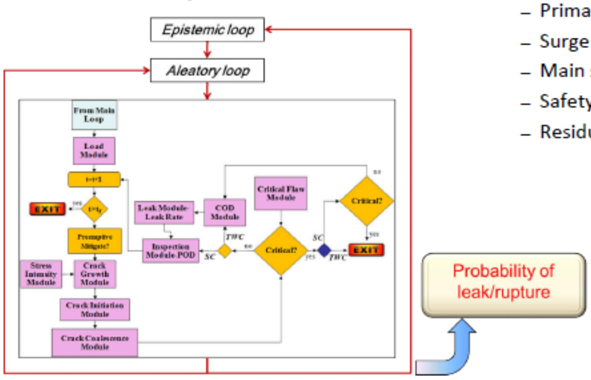
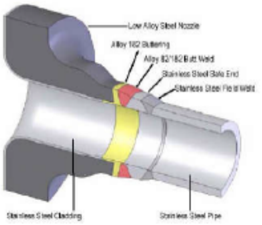
PRO-LOCA - Probabilistic LBB Analysis



- PRO-LOCA code is used for **Leak-Before-Break (LBB)** analysis (criteria: $2c_d < 2c_b$)
- Demonstration that LBLOCA is an extremely rare event
- Active degradation mechanisms:
  - PWSCC for PWR & IGSCC for BWR
  - TMF
- The scope of the LBB analysis includes
  - Primary coolant loop
  - Surge line
  - Main steam line
  - Safety injection line
  - Residual heat removal line





PROACTIVE 2019 - 2021

PROACTIVE is the follow-up project of PROBAB

part	topics	effort
TP-I	Probabilistic Integrity Assessments at critical RPV locations and piping under consideration of active degradation mechanisms (PARTRIDGE, PRO-LOCA, OECD-Benchmark „LBB“)	40 %
TP-II	Experimental Validation of Extended Finite Element Method (XFEM) with respect to application to crack growth simulations. (Experiments and participation at OECD-Benchmark „XFEM“)	40 %
TP-III	Fracture toughness evaluation of RPV-steels by using of small specimen (mini-CT?) (literature study to identify knowledge gaps + definition and start for a PhD-thesis)	20 %

SAFE-II (ENSI, 2015 - 2017)

SP-I: SCC Initiation in Austenitic SS & Ni-Alloys

- PhD thesis of J. Bai: Effect of H_2 on SCC initiation in Alloy 182 under BWR/HWC conditions @ 274 °C
- International MICRIN+ / NUGENIA+ project with 9 partners: Accelerated screening–SSR-Tests (tapered specimens) on SCC initiation (critical strains & stresses) in CW & high Si SS and Alloy 182
- ICG-EAC Round Robin on SCC initiation (constant load tests)
- ECG-COMON Round Robin on ECN & EIS measurements in high-temperature water

SP-II: Environmental Effects on Fracture

- Systematic study & quantification of environmental & hydrogen effects on the fracture behaviour in the upper shelf & brittle to ductile transition region in different RPV steels.
  - DBTT shifts and reduction of resistance to brittle failure due to H and HTW?
  - Major factors of influence & critical system conditions
  - Underlying mechanism, synergies/interactions with other mechanism (EAC, DSA, …)
- PhD thesis (Z. Que) on high-temperature water effects & Post-Doc project (S. Rao) on H effects in air

SP-III: Environmental Effects on Fatigue

- International EU HORIZON2020 INCEFA+ collaborative project, 2015 to 2020 (5 y), 14 partners
  - Effect of mean stress, long static load hold times & surface conditions on fatigue of SS in HTW
  - Development of European EAF analysis procedure, EAF data base (MATDB, JRC)
- PhD thesis (W. Chen) with focus to basic aspects of EAF (mean stress, stress state & mechanism)

LEAD Project (ENSI, 2018 - 2020)

SP-I: SCC Initiation in Austenitic Ni-Alloys & SS (30 %)

- Effect of surface conditions on SCC initiation & surface modification for SCC mitigation, LBB
- New PhD thesis (A. Treichel), 5/2018-4/2022
- H2020 MEACTOS project, 2017- 2020
- ICG-EAC Round Robin on SCC initiation (constant load tests) – phase II

SP-II: Environmental Effects on Fracture (20 %)

- PhD thesis (Z. Que), RPV steels (DSA, S, TE, HAZ), 2/2019

SP-III: Environmental Effects on Fatigue (20 %)

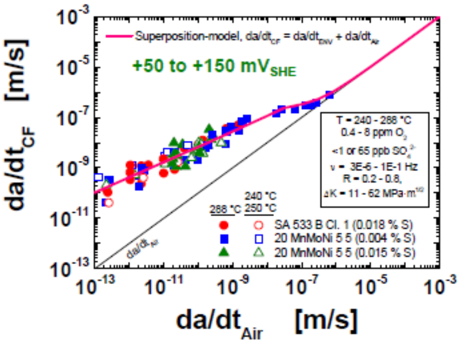
- International EU HORIZON2020 INCEFA+ collaborative project, 2015 to 2020 (5 y), 14 partners
  - Effect of mean stress, long static load hold times & surface conditions on fatigue of SS in HTW
  - Development of European EAF analysis procedure, EAF data base (MATDB, JRC)
- PhD thesis (W. Chen), 3/2020

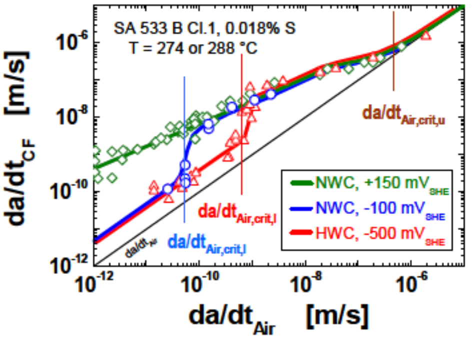
SP-IV: Synergies & Superposition of Ageing Mechanisms (25 %)

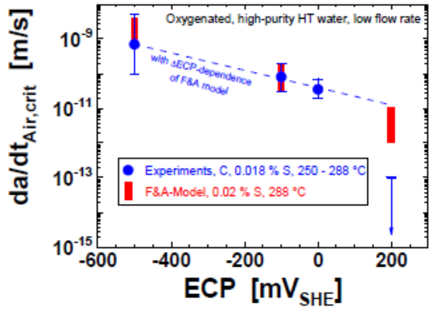
Part-I: Environmental fracture & (IA)SCC of irradiated RPV steels (JRQ)

Part-II: Environmental fracture & SCC of thermally-aged Alloy 182 weld metal (→short range ordering)

PSI Corrosion Fatigue Crack Growth Model







da/dt_{CF} = da/dt_{ENV} + da/dt_{Air}

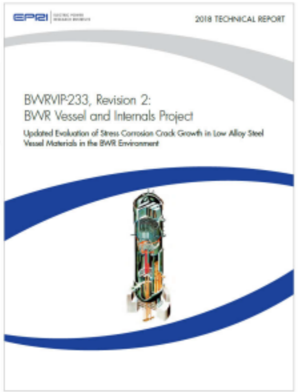
da/dt_{Air} = f (ΔK, R, Δt_R)
= (7.87E-11/ Δt_R) · (ΔK/(2.88-R))$^{3.07}$

da/dt_{Air} < $da/dt_{Air,crit}$ = f (ECP): da/dt_{ENV} = 3-da/dt_{Air}
$da/dt_{Air,\ crit}$ ≤ da/dt_{Air} < 3.7E-8 m/s: da/dt_{ENV} = 6.6E-3 · $(da/dt_{Air})^{0.6}$
da/dt_{Air} ≥ 3.7E-8 m/s: da/dt_{ENV} = 2.3E-7 m/s

$da/dt_{Air,crit}$ for high-sulphur steel and low-flow conditions

ECP = -500 mV_{SHE}: $da/dt_{Air,crit}$ ≈ 7E-10 m/s
ECP = -100 mV_{SHE}: $da/dt_{Air,crit}$ ≈ 8E-11 m/s
ECP = +200 mV_{SHE}: $da/dt_{Air,crit}$ < 1E-13 m/s

BWRVIP-233 R.2 SCC Disposition Lines

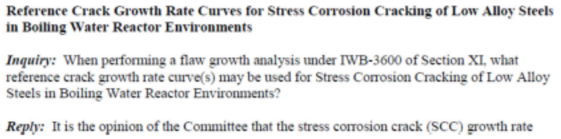


2018 TECHNICAL REPORT

BWRVIP-233, Revision 2:
BWR Vessel and Internals Project

Updated Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment

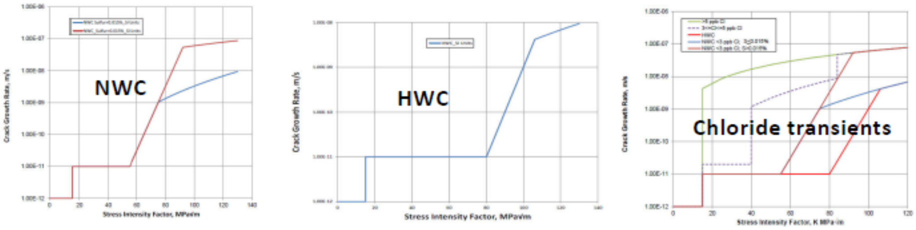
- New SCC disposition lines for RPV steels for BWR/NWC, HWC & chloride transients
- ASME BPV Code Case, positive feedback from all code (sub-) committees, response to US NRC feedback done, final approval expected in May 2019
- Largely based on PSI results and suggestions
- Adjustment of AL 1 of EPRI BWR WC Guidelines for chloride to 3 ppb in 2016



Reference Crack Growth Rate Curves for Stress Corrosion Cracking of Low Alloy Steels in Boiling Water Reactor Environments

Inquiry: When performing a flaw growth analysis under IWB-3600 of Section XI, what reference crack growth rate curve(s) may be used for Stress Corrosion Cracking of Low Alloy Steels in Boiling Water Reactor Environments?

Reply: It is the opinion of the Committee that the stress corrosion crack (SCC) growth rate



PSI support to new ASME Code Cases for EAC

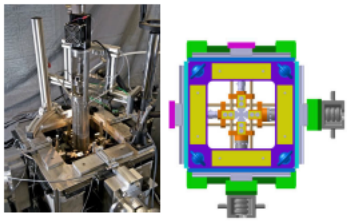
- Code Case N-XXX
  “Reference Crack Growth Rate Curves for SCC of Low Alloy Steels in BWR Environments”
  Draft close to final approval,
  (C&S Connect Record # 17-3016)
- Two new Code Cases for BWR Fatigue Crack Growth Rates are under preparation
  - for Low Alloy Steel (i.e., BWR Version of N-643-2) expected to use data provided by PSI (C&S Connect Record # 19-5)
  - for Stainless Steel (i.e., BWR Version of N-809) (C&S Connect Record # 19-6)

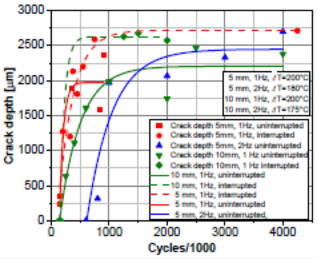
Non-Standard Fatigue Tests for Model Validation & Development

(supported by swissnuclear)

TMF

with biaxial pre-loading

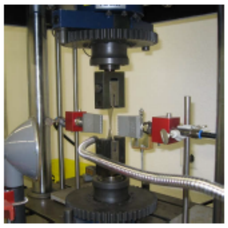


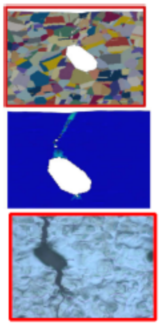


- Crack arrest at 2 to 2.5 mm = f (frequency)
- Strong effect of surface conditions on initiation

HCF

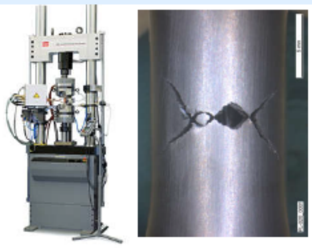
with micro notch (FIB)

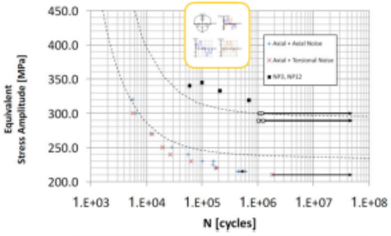




Multiaxial HCF

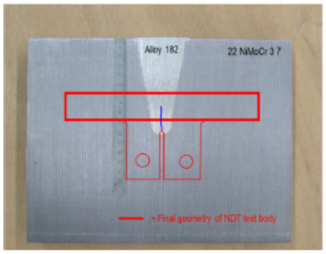
tension-torsion, proportional & non-proportional

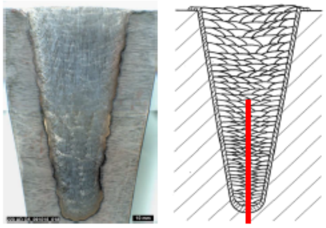




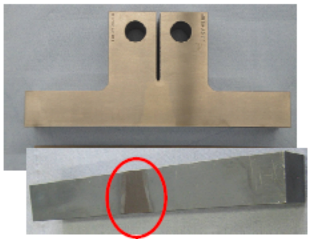
Significant effect of small non-proportional noise on fatigue life!

Swiss contribution to PARENT, PIONIC

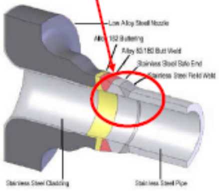




1. IG SCC crack (BWR/NWC) in middle of bulk weld metal ~ 20 % of wall thickness



2. Detailed UT characterization by SVTI
3. Cutting of weld segment & insert of segment in weld mock-up by minimally invasive welding

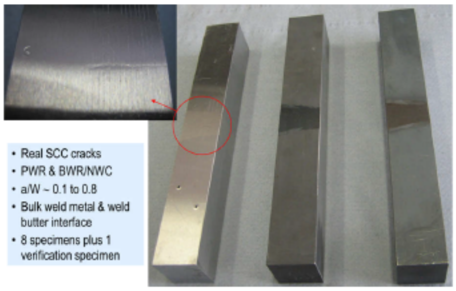


PARENT (2012-16): NDT Test Bodies with SCC Cracks

Detection and, in particular, sizing of SCC defects in DMWs represents a challenge and is related to relevant uncertainties. Crack depth is often significantly underestimated by NDT!

PARENT: Program to Assess the Reliability of Emerging Nondestructive Techniques
follow-on project to **PINC: Program for the Inspection of Nickel Alloy Components**

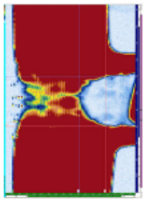
- Participation of Swiss consortium (ENSI, PSI, ALSTOM, SVTI, EMPA) in PARENT-Project
- International program including regulators, industrial groups and research institutions
- Assessment & quantification of established & new promising NDE techniques
- NDT tests bodies with well characterized SCC cracks for open round robin as PSI contribution
- Participation in open and closed round robin programs (ALSTOM, SVTI)



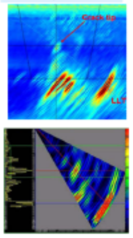
- Real SCC cracks
- PWR & BWR/NWC
- a/W ~ 0.1 to 0.8
- Bulk weld metal & weld butter interface
- 8 specimens plus 1 verification specimen

BWR/NWC SCC Crack

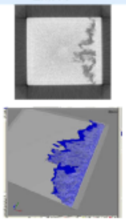
UT immersion
Alstom



UT phased array
SVTI, Alstom



X-ray radiography & tomography
EMPA



PIONIC

Background:

- Optimization of NDT methods for SCC detection in DMW and Ni-alloys
- Follow-up project of PARENT & PINC

Administrative issues:

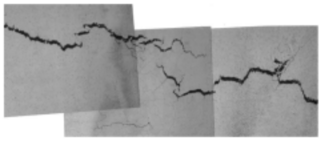
- Contract of ENSI with US NRC (in-kind contributions), contract of PSI with ENSI

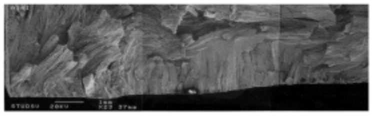
PSI Tasks:

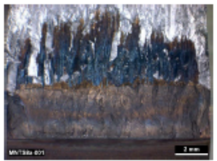
- 2 (or 3) NDT specimens with real IG/ID SCC with short cracks (~ 20 % of wall thickness), NWC with chloride and high-purity water in final phase, interrupted with UT at Alstom & SVTI?
- 2 x 1T C(T) specimens for quantitative crack configuration characterizations: Crack openings, branching, un-cracked ligaments, surface roughness in loaded and unloaded conditions
- ~ 1 year of total testing time

Status:

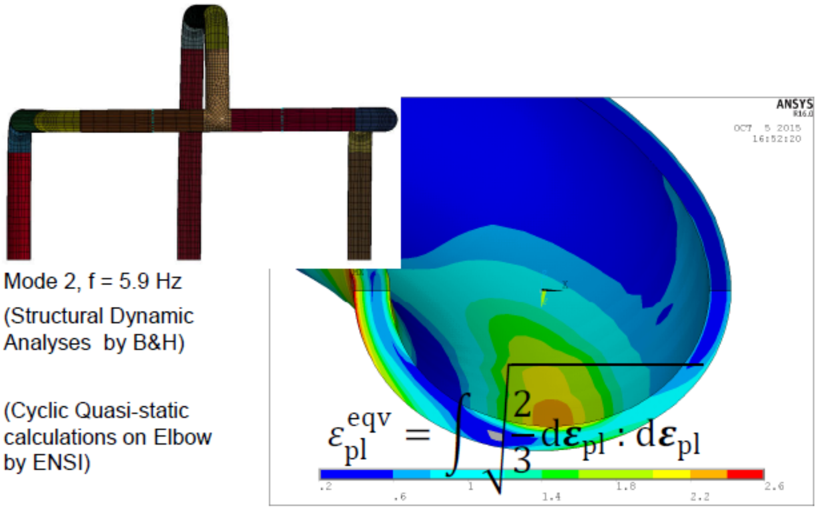
- Specimens fabricated and pre-cracked, 3rd test is running,
- NDT pre-characterization of initial conditions at SVTI in December 2018







Research activities, OECD/NEA MECOS Benchmark



$$\varepsilon_{pl}^{eqv} = \int \sqrt{\frac{2}{3} d\boldsymbol{\varepsilon}_{pl} : d\boldsymbol{\varepsilon}_{pl}}$$

Mode 2, f = 5.9 Hz

(Structural Dynamic Analyses by B&H)

(Cyclic Quasi-static calculations on Elbow by ENSI)

Results of Fracture Mechanics RPV weld project

- Inclusion of MC-Concept to the RegGuide ENSI-B01, App. 5
- An Equation to convert T_0 (ASTM E-1921) to RT_{ref} was developed, which have to be used instead ASME CC-N-629 or N-631
- ENSI-B01, Method II:

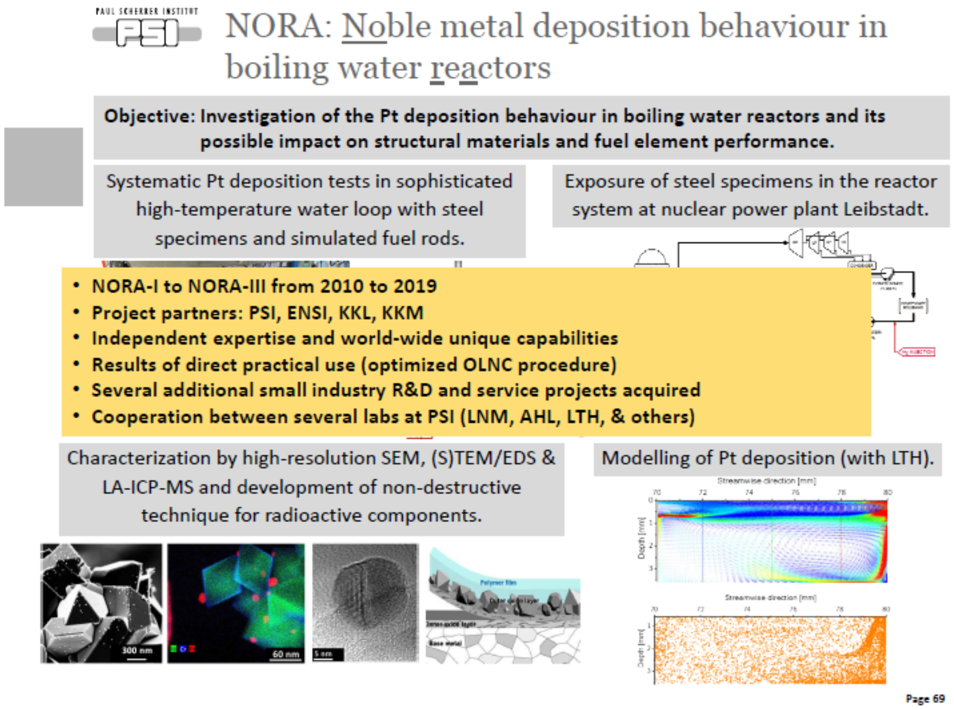
$$RT_{ref} = T_0 + 14.4K + \Delta T_S + \sqrt{324/n + 16 + \Delta T_M^2 + \Delta T_T^2}$$

T_o	reference temperature acc. to ASTM E 1921 (MC approach)
n	number of valid tests
ΔT_s	=0 for 1T-C(T) specimens; =10K for 0.4T-SEN(B) specimens
ΔT_M	=0 for base material; =6K for weld material
ΔT_T	=0 for 1T-C(T) specimens; =5K for 0.4T-SEN(B) specimens

- ASME CC-N-629, N-631:

$RT_{T0(}{}^{\circ}C) = T_0 + 19.4K$

$RT_{T_0}(°F) = [T_o + 35](°F)$



PAUL SCHERRER INSTITUT

NORA: Noble metal deposition behaviour in boiling water reactors

Objective: Investigation of the Pt deposition behaviour in boiling water reactors and its possible impact on structural materials and fuel element performance.

Systematic Pt deposition tests in sophisticated high-temperature water loop with steel specimens and simulated fuel rods.

Exposure of steel specimens in the reactor system at nuclear power plant Leibstadt.

- **NORA-I to NORA-III from 2010 to 2019**
- **Project partners: PSI, ENSI, KKL, KKM**
- **Independent expertise and world-wide unique capabilities**
- **Results of direct practical use (optimized OLNC procedure)**
- **Several additional small industry R&D and service projects acquired**
- **Cooperation between several labs at PSI (LNM, AHL, LTH, & others)**

Characterization by high-resolution SEM, (S)TEM/EDS & LA-ICP-MS and development of non-destructive technique for radioactive components.

Modelling of Pt deposition (with LTH).

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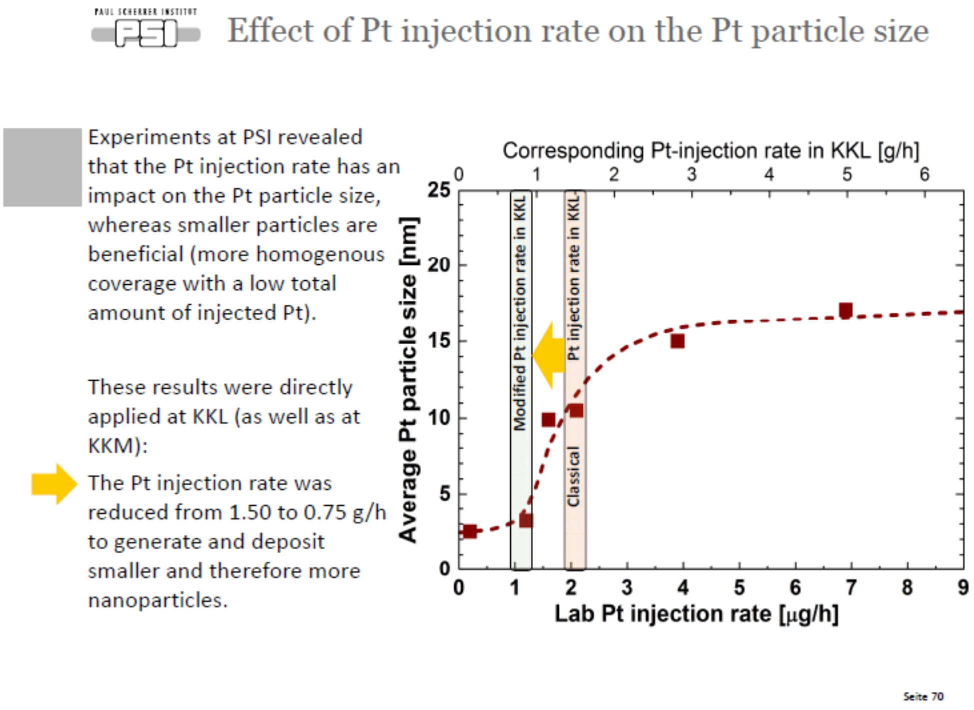

PAUL SCHERRER INSTITUT

Effect of Pt injection rate on the Pt particle size

Experiments at PSI revealed that the Pt injection rate has an impact on the Pt particle size, whereas smaller particles are beneficial (more homogenous coverage with a low total amount of injected Pt).

These results were directly applied at KKL (as well as at KKM):

The Pt injection rate was reduced from 1.50 to 0.75 g/h to generate and deposit smaller and therefore more nanoparticles.

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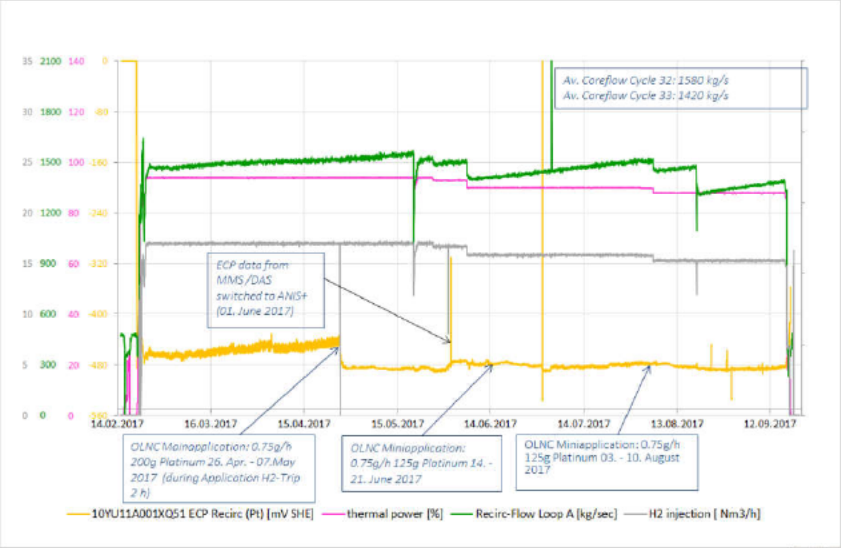
HWC/OLNC: ECP-run Cycle 33



Av. Coreflow Cycle 32: 1580 kg/s
Av. Coreflow Cycle 33: 1420 kg/s

ECP data from MMS /DAS switched to ANIS+ (01. June 2017)

OLNC Mainapplication: 0.75g/h 200g Platinum 26. Apr. - 07.May 2017 (during Application H2-Trip 2 h)

OLNC Miniapplication: 0.75g/h 125g Platinum 14. - 21. June 2017

OLNC Miniapplication: 0.75g/h 125g Platinum 03. - 10. August 2017

Summary

- In Switzerland the establishing of a systematic ageing management started in 1991
- From ENSI`s perspective, a comprehensive and effective ageing management is implemented for the RPVs in the Swiss nuclear power plants, with which relevant ageing effects are detected at an early stage and effective corrective actions are initiated.
- Within the AMP all relevant ageing mechanisms for the RPV components are identified and evaluated according to the current state-of-the-art of science and technology.
- Based on results of the extensive inspection programmes, specific corrective actions have been taken for identified ageing-related damage and degradation.
- With the existing AMP for the RPVs, safe long-term operation for an operating period of 60 years, or for KKM until the final shutdown in 2019, is ensured.

Schweizerische Eidgenossenschaft
Confédération suisse
Confederazione Svizzera
Confederaziun svizra

Eidgenössisches Nuklearsicherheitsinspektorat ENSI

Acknowledgement

Special thanks to

Hans-Peter Seifert and colleagues (PSI)
Jens Heldt (KKL)
and to all colleagues from ENSI
supported this presentation

and for providing material.

Thank You for your kind attention.

NRC International Workshop on Age-Related Degradation of RPV and Internals
Ralph Döring / ENSI (Switzerland), Rockville, MD, 23-24 May 2019

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B.24 Operating Experience of a Swiss BWR (J. Heldt)

Operating experience of a Swiss BWR (KKL)

Jens Heldt
Leibstadt NPP

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Kernkraftwerk Leibstadt

Field Experience Leibstadt NPP

Contents

Introduction

Leibstadt Nuclear Power Plant (KKL)

- BWR/6, 1275 MWe

  EPU of 114% accomplished in 2003

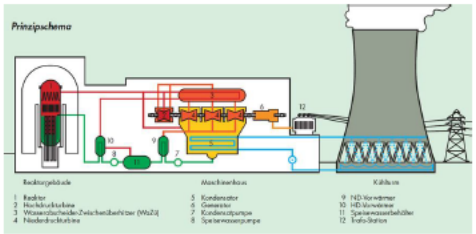
  \+ LP turbine refurbishment in 2010

  \+ new turbine generator in 2012

  \+ new moisture separator reheater in 2017

  \+ new condenser in 2020

  ….
- Mark III containment
- Commercial operations: 1984
- Newest and largest NPP in Switzerland
- Located at the River Rhine on the border between Switzerland & Germany



EAC in Reactor Water

KKL EAC in Reactor Water

Findings

Three different kinds of indications so far:

- Recirculation loop piping
- Core shroud (horizontal weld)
- Dissimilar Metal Weld (DMW) of feed water nozzle to safe end

---

Recirculation Loop Piping

- Made out of 316 NG

  low carbon content < 0.02 % (from 0.007 to 0.018%)

- Most of all indications found at shop welds made by Hitachi

  acceptable according ASME Code Sect. XI

- Augmented inspections by UT
- All indications did not show growth since 2001

  in accordance with the observations made for Japanese recirc. piping*

* = e.g. *K.Kumagai et.al. Proceedings of ASME-PVP 2004: 2004 ASME/JSME Pressure Vessels and Piping Conference San Diego, California, July 25 – 29, 2004*

Core Shroud

- Made out of 304L / 308L
- OD of Weld SD-007: two small indications found by VT in 2012
  - maximum length of 22 mm
- Reinspection after 2 und 5 years revealed no apparent change of both indications

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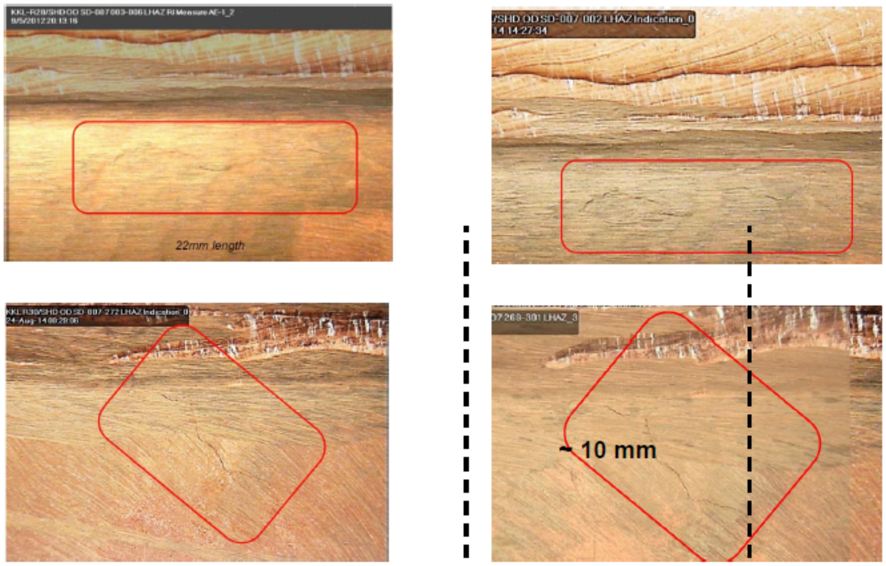
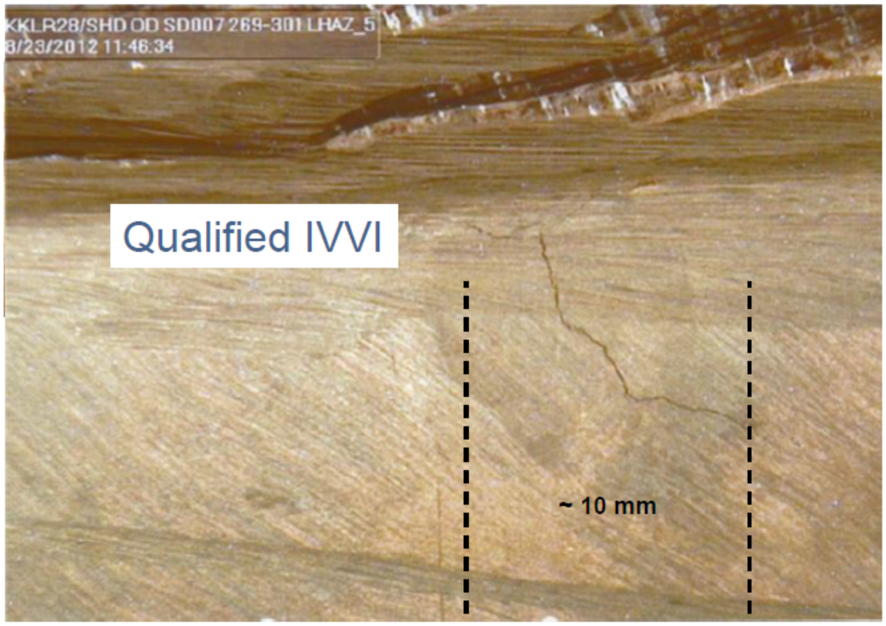
Indications in Core Shroud

2012

2014



Indications in Core Shroud



SCC of DMW

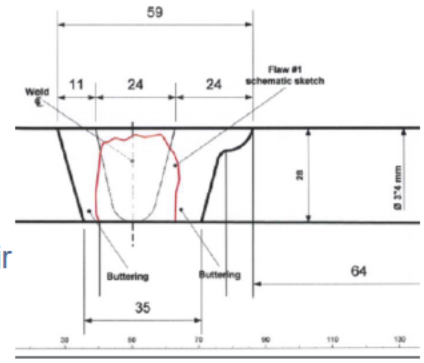
DMW of Feed Water Nozzle to Safe End

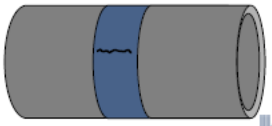
- Ferritic nozzle and safe end
- DMW: Alloy 182 (Butter) / Alloy 82
- No Mechanical Stress Improvement (MSIP)
- ID-connected axial indication found in 2012
- Reevaluation of prior UT-inspection in 2004
  - ~1.5 mm/year crack growth
- Full Structural WOL with Alloy 52

RPV Nozzle Dissimilar Metal Welds (DMW)

Outage 2012: Indication overlaid at N5-nozzle

- Axial orientation
- Found in weld metal
- Length 22 mm (*shorter at base?)*
- Depth 26 mm (~ 93% wall thickness)
- Multifaceted and branched
- Emanating from edge of ID weld repair
- Clear ID connection
- Flaw is indicative of interdendritic stress corrosion cracking





Axial orientation (schematic)

After WOL Repair: Additional Laboratory Tests

Material Alloy 52 tested with the identical:

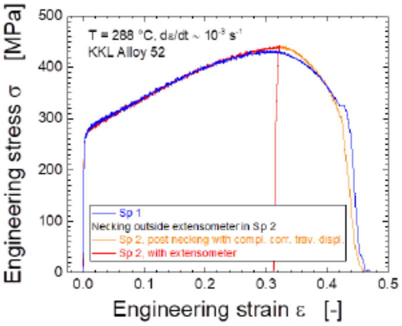
- Chemical composition
- Welding parameters

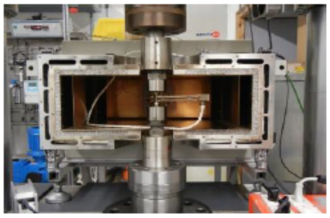
Test conducted:

- Elevated temperature tension tests in air
- Fracture toughness tests in air
- SCC tests under simulated BWR-conditions

Elevated Temperature Tension Tests

Parameter	Sp. #1	Sp. #2	Mean value
E [GPa]	171	167	169
$YS_{0.2}$ [MPa]	260	262	261
UTS [MPa]	430	440	435
FS = (UTS+YS)/2 [MPa]	345	351	348
UE [%]	31	32.5	31.8
EF [%]	45.4	45.7	45.6
RA [%]	77.2	82.9	80.1

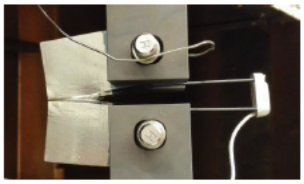




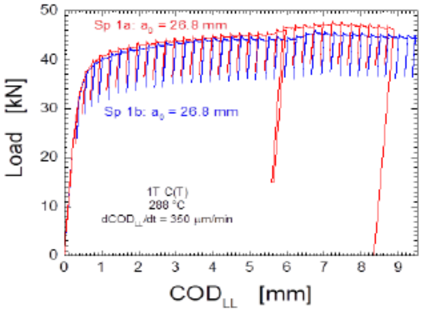
After WOL Repair: Additional Laboratory Tests

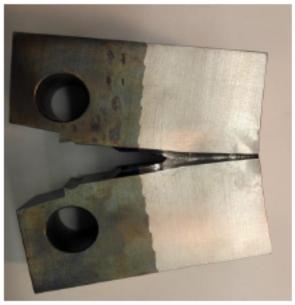
Fracture Toughness Tests according to ASTM E1820-13

Sp	Type	a_0 [mm]	Δa [mm]	Δa_{ULC} [mm]	J_Q [kJ/m^2]
2b	0.5T C(T)	12.37	3.06	3.10*	1089**
2c	0.5T C(T)	13.20	1.30	1.30	1118
2d	0.5T C(T)	13.35	2.65	1.97	948
1a	1T C(T)	26.80	2.80	2.81	1420
1b	1T C(T)	26.80	4.10	3.53	914

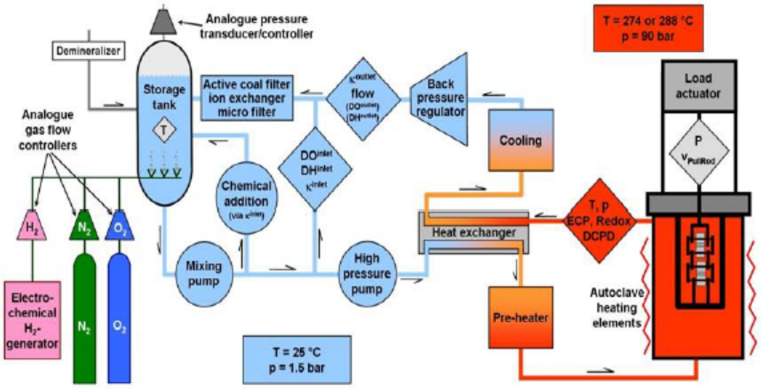


High toughness confirmed

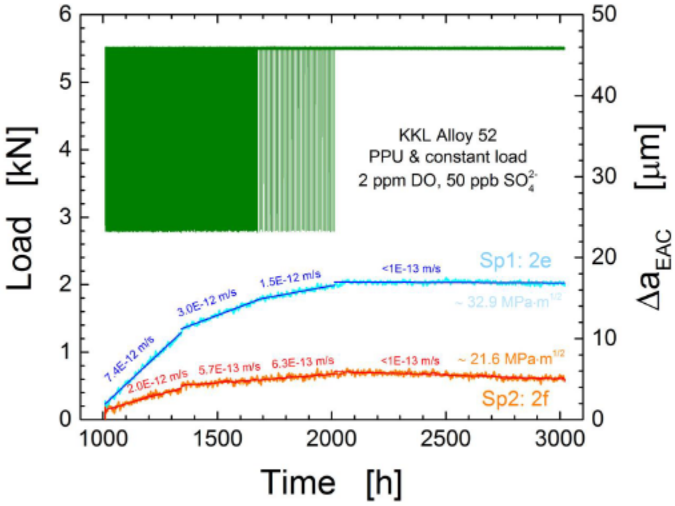




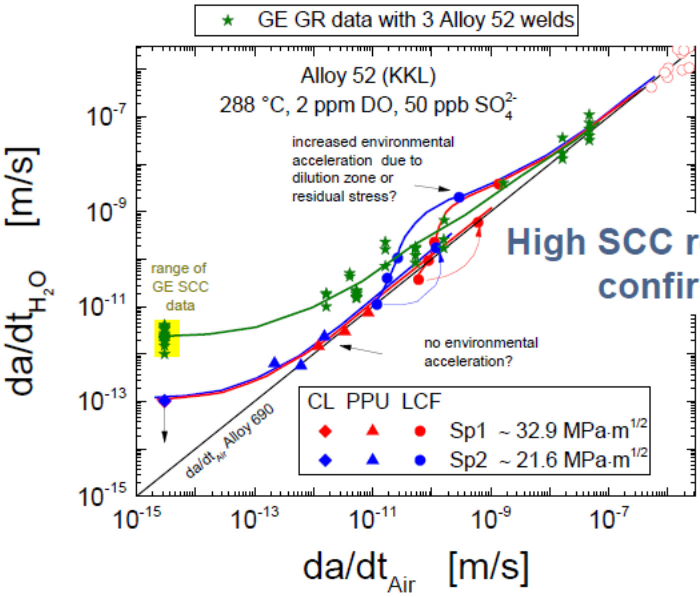
SCC Tests in a high-temperature water loop



SCC Tests in a high-temperature water loop



SSC Tests in a High-Temperature Water Loop



High SCC resistance confirmed

After WOL Repair: Analysis on SCC Susceptibility

- **Assessment of all RPV DMW's**
  - + Operational experience
  - + OLNC / HWC mitigation
  - + ID-repairs of DMW's
  - + Operational loads at DMW's
    - → supported by a fracture mechanics analysis
- **Provide qualification of a MSIP pilot application to the Swiss regulator**

RPV Nozzle DMW

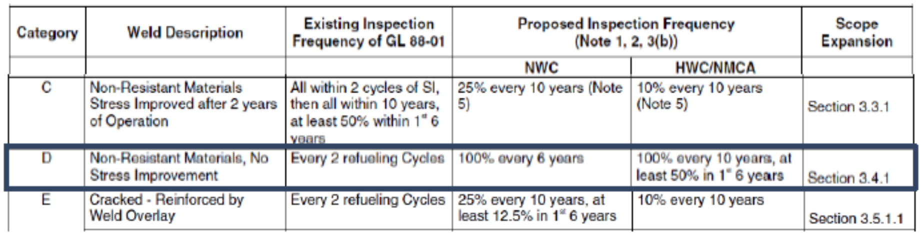
DMW: Alloy 182 (Butter) / Alloy 82

	Materials		
	Nozzle	**Safe End (SF)**	**SF to Safe End Extension (SFE)**
N2 Recirculation Outlet	SA-508 CL2	SA-336 CL F8	-
N3 Recirculation Inlet	SA-508 CL2	SB-166	SA-336 CL F8
N5 Feedwater	SA-508 CL2	SA-508 CL 1	-
N6 Core Spray	SA-508 CL2	SB-166	SA-508 CL 1
N7 RHR/LPCI	SA-508 CL2	SB-166	SA-508 CL 1
N10 JP Instrument.	SA-508 CL2	SA-336 F8	-
N11 CRD Return	SA-508 CL2	SA-336 F8	-

BWRVIP-75-A

"Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules"

- Re-assessment of Inspection schedules required by Generic Letter 88-01
- BWRVIP-75-A issued in 2005 and accepted by USNRC



Category	Weld Description	Existing Inspection Frequency of GL 88-01	Proposed Inspection Frequency (Note 1, 2, 3(b))		Scope Expansion
			NWC	HWC/NMCA	
C	Non-Resistant Materials Stress Improved after 2 years of Operation	All within 2 cycles of SI, then all within 10 years, at least 50% within 1st 6 years	25% every 10 years (Note 5)	10% every 10 years (Note 5)	Section 3.3.1
D	Non-Resistant Materials, No Stress Improvement	Every 2 refueling Cycles	100% every 6 years	100% every 10 years, at least 50% in 1st 6 years	Section 3.4.1
E	Cracked - Reinforced by Weld Overlay	Every 2 refueling Cycles	25% every 10 years, at least 12.5% in 1st 6 years	10% every 10 years	Section 3.5.1.1

Assessment of Operational Experience

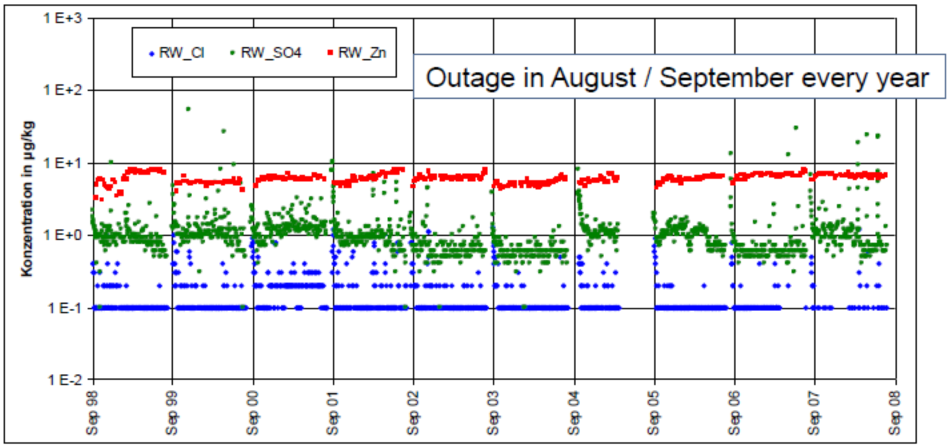
since 2005 to complete information found in BWRVIP-75-A

- **Information was hard to compile:**
  - different lists, private communications etc.
  - part of information missing (category C / D, inspection history)
- **Input and assistance from EPRI (C. Wirtz) and additional input from utilities** (provided by Sierra Technologies)
- **Assessment by KKL**
  - Crack initiation in Alloy 182 but also crack growth in Alloy 182
  - No (significant) crack growth in base material (nozzle, SF and SFE)
  - **Very few indications with > 0.75 t since 2005**

Most important: encoded qualified UT-inspection method

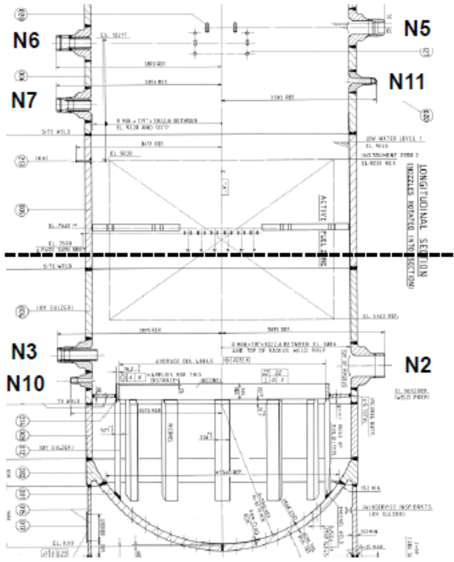
Water Chemistry

Low concentration of chloride and sulfate



Assessment of OLNC / HWC Mitigation

according to BWRVIP-219



Not mitigated:

- N5: Feed Water
- N6: Core Spray (HPCS / LPCS)
- N7: RHR / LPCI
- N11: CRD-Return

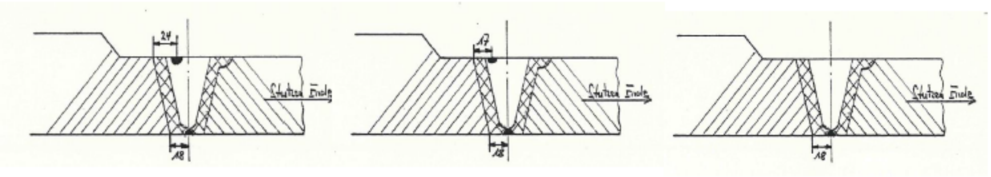
Mitigated:

- N2: Recirc Out
- N3: Recirc Inlet
- N10: Jet Pump Instrumentation

KKL-RDB, Ausschnitt aus Zeichnung Z06536

Assessment of ID-Repairs

All N5 nozzle to safe end DMW have ID-repairs



30°, max. depth of 5 mm

150°, max. depth of 4 mm

90°, depth of 3 mm

Two other ID-repairs found:

- N2 SF
- N7 SFE (shop weld)

Operational Stresses

E.g.: hoop stresses σ_h (axial crack growth)

Nozzle	DMW	Outer Dia [mm]	Thickness [mm]	σ_h [MPa]
N2	SF	600	46	57
N3	SF	358	31	49
N5	SF	374	28	58
N6	SF	342	33	45
	SFE	335	23	**65**
N7	SF	354	30	51
	SFE	335	23	**65**
N10	SF	140	21	**27**
N11	SF	136	21	**25**

Qualitative Assessment

Important Factors:

Operational experience	BWRVIP-75A inspection frequencies
Water chemistry	good practice at KKL
ID-repairs	impact on initiation and growth of SCC
Operational stresses	for SFE > SF («wall thickness»)
Qualified UT-technique	very important

Approach:

BWRVIP-75A as a «base line»

Shorter UT-intervalls for welds: (a) with ID-repair

(b) SFE (if not mitigated by HWC)

Fracture Mechanics Analysis

Calculation of «residual live time»

- Welding Residual Stresses (FE-Analysis*) and operational stresses
- Crack growth by SCC (>> Fatigue CGR)
- Circumferential and axial crack growth
  axial cracks: «natural flaw growth»; crack growth in Alloy 182/82
  circumferential cracks: crack growth in SS and in Alloy 182/82
- Crack growth rates according to BWRVIP-59-A / BWRVIP-114-A
- Acceptable flaw dimensions according ASME Sect. XI, Appendix C
- Initial crack depth: 10% of wall thickness

* As described in:
D. Sommerville et al., Simplified Dissimilar Metal Weld Through-Wall Weld Residual Stress Models for Single V Groove Welds in Cylindrical Components, Paper No. PVP2014-28828ASME, 2014 Pressure Vessels and Piping Conference

Shortened UT-Inspections Intervals

DMW	Inspection [y]	Remarks
N2 Recirc.Outlet SF	10 / 5	Mitigated by HWC 5 years interval because of ID-repair at one weld
N5 Feedwater SF	3	Not mitigated by HWC ID repairs
N6 Core Spray SFE	4	Not mitigated by HWC
N7 RHR/LPCI SFE	4 3	Not mitigated by HWC 3 years for one weld with ID-repair

Concluding Remarks

Concluding remarks

Important for the assessment of SCC EAC / SCC

- Understanding of the mechanisms / phenomena
- Disposition lines for crack growth

- Operational Experience
- Fabrication History

- NDE capability

Thank you very much for your attention

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